DIVERSITY IN NUCLEAR

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8th - 14th March

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International Youth Nuclear Congress 2020

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WELCOME WORDS

Dear readers,

The International Youth Nuclear Congress (IYNC) was held in Sydney, Australia on 8-13 March 2020, and co-hosted with the Australian Young Generation in Nuclear (AusYGN). IYNC and AusYGN are organisations that are committed to ensuring that the youth are engaged and supported within the nuclear industry, and able to capitalise on the numerous professional opportunities for careers, networking and development. Under the theme Diversity in Nuclear, the mission of the IYNC2020 was to promote and enable the diversity of people engaged in the many peaceful uses of nuclear science and technology.

IYNC2020 aimed to promote and encourage diversity of people. Diversity comes in many forms, including gender, culture, educational background, professional experience and geographical location. It successfully showcased the diversity in the peaceful uses and applications of nuclear science and technology by encouraging interactions between participants, particularly in the sharing of knowledge and ideas between professionals of different personal and professional backgrounds and different generations of nuclear experts.

An important aspect of the IYNC has always been its technical contributions. IYNC is the perfect platform for young professionals to share their latest works and research with other participants from all around the world. Holding true to its theme, the IYNC2020 had a variety of technical offerings to delegates including keynote sessions, plenaries, panels workshops, technical presentations and posters. Topics ranged from diagnosing cancer to improving agricultural production all presenting nuclear science and technology as a critical contributor to global sustainable development goals.

This event was the first of its kind in Australia and not only showcased the breadth of nuclear applications, it celebrated the diversity of people who deliver these outcomes for society.

It is our hope that you enjoy this book,

Luca Capriotti
General Co-Chair IYNC2020

Alex Borovskis
General Co-Chair IYNC2020

Technical Track Managers Acknowledgments

IYNC would like to thank to all the members of Technical Tracks Team for their hard work and commitment to our shared goal.

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Diverse Reactor Emergency Cooldown System (Project AES-2006 Export)

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I. Introduction

The safety theory of nuclear power plants is going through a period of its formation. If its General concepts have been known for a relatively long time, then the issues of analysis and synthesis of safety by various criteria, methods of forecasting, localization and prevention of accidents, generalization of knowledge about specific accidents are only being formed [1].

Directly the goals and principles of NPP safety are disclosed in a number of IAEA documents. It is considered that the goals and principles themselves are not normative requirements, but national safety regulations should reflect the goals and principles in the form of specific requirements.

Russia is developing a safety concept for advanced nuclear power units (NPP-2006 Export project). The NPP-2006 Export project is based on the use of thermal neutron reactors with pressurized water (WWER-1000).

The design of NPP-2006 Export takes into account the requirements of Russian regulatory documents in the field of NPP design, as well as international and other national documents such as:

- IAEA recommendations and standards;
- requirements of European operating organizations for new generation nuclear power plant projects with LWR (EUR) reactors);
- WENRA requirements.

Since a nuclear reactor is a source of increased danger, and the safety of any NPP can not be absolute, the safety concept of perspective NPP is a set of modern scientific and technical principles, the use of which allows you to optimally meet regulatory requirements and provide a significant reduction in the risk of operation of perspective power units in comparison with existing reactor facilities.

In the safety concept of international NPP developed by JSC ATOMPROEKT (NPP-2006 Export), analyses are carried out to determine the necessity and effectiveness of the use of engineering and technological means, or accident management procedures for modes with complex failure sequences, which include failures in excess of those considered in the analysis of design accidents (including failures due to a common cause), but do not lead to fuel melting. These modes are called design extension conditions (DEC).

To minimize the probability of such accidents occurring, it is necessary to use diverse safety systems to improve the NPP safety. One of these diverse systems is the reactor emergency cooldown system.

II. Safety Consept for Project NPP-2006 Export

The main goal of the safety concept of the NPP-2006 Export Project is to create at each level of the defense in depth (DiD) [2] an independent engineering mean to ensure the three fundamental safety functions:

- reactivity control;
- cooling of the reactor core;
- localization and reliable retention of radioactive products.

The main task of cooling the reactor core is to prevent the destruction of fuel rods due to their overheating. Therefore, in all modes of operation of a nuclear reactor, it is necessary to maintain the correspondence of the amount of heat generated in the core and removed from it by the heat removal systems.

During normal operation, the heat is removed by the primary circuit coolant, transferred through the steam generator to the second-circuit coolant, and diverted to the final absorber using cooling towers or spray pools.

For emergency modes, safety systems that ensure heat removal from the core. Such as a system of passive heat removal through steam generators, low and high pressure zone emergency cooling systems, the system of removal of residual heat from the reactor plant.

For severe accidents without fuel melting in case of complete loss of the function of heat removal from the reactor plant through the standard systems of residual heat removal and cooling through the first circuit, a new diverse system of reactor emergency cooldown was developed in the NPP-2006 Export project.

The loss of the heat removal function from the reactor plant can occur in the DEC under the following initial events:

- Station Blackout;

- Small leak with failure of high and low pressure ECCS (active part);
- Long-term loss of heat removal by scheduled and emergency cooldown systems with the reactor pressure vessel head removed and/or reactor sealed;
- Crash of a military or civil aircraft into the safety building;
- Common cause failure of the intermediate circuit;
- Complete failure of the normal residual heat removal system.

The initial events under consideration that do not result in fuel melting are potentially more severe than design accidents. The frequency of occurrence of these events is less than 10^{-6} 1/year.

III. Diverse Reactor Emergency Cooldown System

This diverse system is designed to cooldown the reactor plant at the water-water stage in the event of complete loss of the function of heat removal from the reactor through the standard means of the residual heat removal system and cooling through the primary circuit. This system works in the conditions of DEC and performs the following functions:

- Long-term removal of residual heat from the reactor plant after shutdown;
- Maintenance of primary coolant reserve in case of small leakages (injection into the primary circuit under low pressure);
- Increase in the concentration of boric acid in the primary circuit system to ensure subcriticality of the core in “cold” state;
- Makeup of spent fuel pool;
- Removal of heat from the molten core.

The system performance is provided by the following systems:

- primary circuit coolant system;
- standby system of technical water for responsible consumers using a fan cooling tower;
- reliable power supply system DEC;
- instrumentation and control system for DEC (Diverse protection system).

The connection of the system to the primary circuit is organized on the non-disconnected part of the pipelines of the standard systems for removing residual heat from the reactor plant in the reactor building (UJA) (see figure 1). The main equipment of the reactor emergency cooldown system (pumps, heat exchangers) is installed in the auxiliary building (UKA).

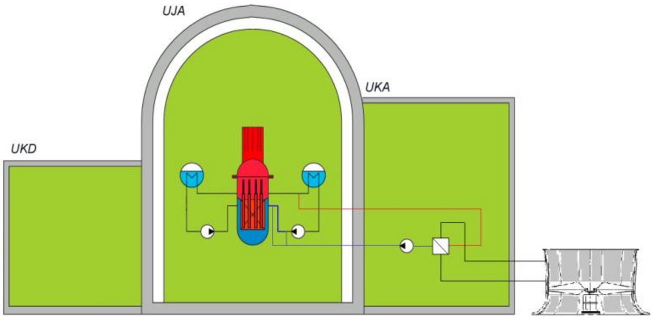

Figure 1. Layout of the reactor emergency cooldown system.

Reactor emergency cooldown system consists of two identical and completely independent trains. Each train includes the following:

- reactor emergency cooldown pump;
- spent fuel pool pump;
- reactor emergency cooldown heat exchanger;
- pipelines;
- valves.

Schematic diagram of diverse reactor emergency cooldown system is presented on the Figure 2.

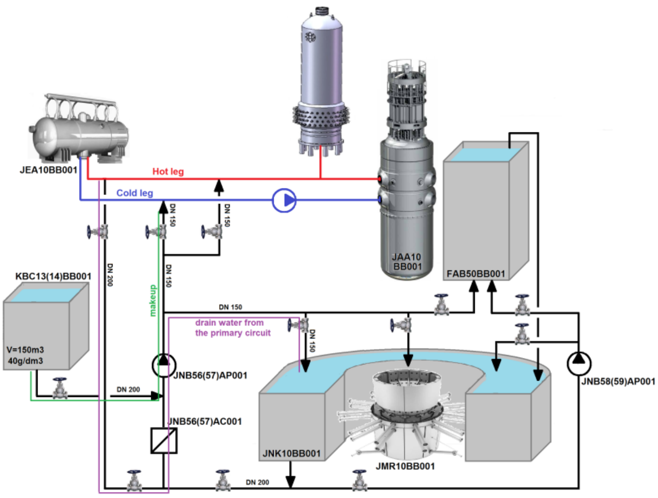

Figure 2. Schematic diagram of diverse reactor emergency cooldown system.

IV. System Operation

In case of loss of the function of heat removal from the reactor the standard safety systems the diverse system operates in a closed circuit: the reactor – the residual heat removal system JNA – pipeline diverse system – the reactor emergency cooldown heat exchanger JNB56(57)AC001 – the reactor emergency cooldown pump JNB56(57)AP001 – pressure pipe diverse system - the low pressure safety injection system JNG - reactor. At reaching of reactor temperature 60 °C, the system provides residual heat removal from the reactor core and maintenance of temperature in the primary circuit 60 °C at most.

In case of loss of function of cooling of the spent fuel pool by standard systems the diverse system ensures recharge of the spent fuel pool. The spent fuel pool emergency recharge pump JNB58(59)AP001 supplies low-concentration borating water to the spent fuel pool from the containment sump tank JNK10(40)BB001.

In the modes associated with a small leak of the coolant of the primary circuit with a failure for a common reason of safety systems, the pump JNB56(57)AP001 provides emergency makeup of the primary circuit system from the containment sump tank JNK10(40)BB001.

If it is necessary to borating the primary circuit in the DEC modes, the diverse system makeup of the primary circuit. The makeup is carried out by the JNB56(57)AP001 pump from the high concentration of boric acid storage tank KBC13(14)BB001 in the hot or cold leg of the reactor. The purge is carried out by gravity from the reactor through the heat exchanger JNB56(57)AC001 in the containment sump tank JNK10(40)BB001.

In severe accidents with fuel melting, this diverse system can be used to cooldown the core catcher. Cooling of the core catcher is organized by the following circuit: containment sump JNK10(40)BB001 - heat exchanger JNB56(57)AC001 - pump JNB56(57)AP001 – core catcher JMR10BB001 - containment sump tank JNK10(40)BB001.

Conclusion

This engineering mean was designed for achieve the safety state of NPP after several accident for AES-2006E project. For the new reactor emergency cooldown system will be carry out a calculation justification of all operation modes in order to fulfill acceptance criteria for several accidents (DEC).

The developed new diverыy system in the NPP-2006 Export project minimizes the probability of accidents with radiological consequences and makes it as competitive on the international market.

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Automatic control system for NPP participation in electrical current frequency regulation in the power network with thermal accumulators

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I. INTRODUCTION

Increasing the number of NPPs in total electricity production and a large equipment degradation of thermal power plants attracted to work in variable operating modes, raises the question of attraction NPPs for regulation the power and electrical current frequency. Now in Russia NPPs work in stationary base mode (in other words, operated at a constant power).

Currently, according to the legislation of Russian Federation, all generating facilities (except for NPPs with reactors type fast breeder reactor (BN type) and high-power pressure-tube reactor (RBMK type)) shall participate in the regulation of the electrical current frequency in the network [1].

Today technical problem in Russia is the absence of NPP participation in the regulation of electrical current frequency in the power network due to low maneuverable characteristics of the reactor cores.

II. THERMAL ENERGY STORAGE SYSTEM

One of solutions of this problem is achieved by using Thermal Energy Storage System (TESS) at the NPP with reactor type VVER [2].

Today in Russia TESS is regarded as the most cost-effective option to participate NPPs only in the Operations load schedule without changing of the reactor power [2].

One of the possible variants of the thermal scheme of a nuclear power plant with thermal energy storage system is shown on Fig. 1 (the TESS is inside of the dark frame).

In this variant oil is used as an accumulating substance – TLV-330. TESS has two modes of operation: charging mode and discharging mode.

Charging mode of the TESS - the heating of the oil is realized by the heat of condensation of fresh steam portion in the special oil charge heat exchanger. The portion of fresh steam is taken from the main stream from steam generator at the time of NPP power reduction (for example, at night time). Heated oil is accumulated in the special storage tank.

In the Discharging mode of the TESS, at the time of NPP power increase in day time, heat recovery is realized in the special oil discharge heat exchangers, which connected to the regeneration system of the turbine.

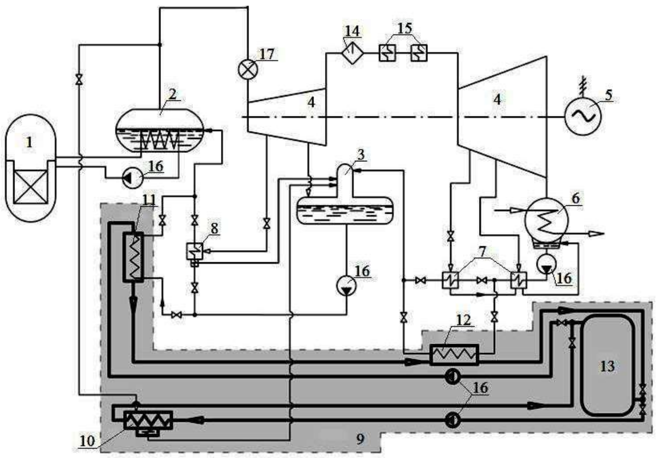

Figure 1. Possible variants of the thermal scheme of a nuclear power plant with thermal energy storage system. In this diagram is indicated: 1 – reactor; 2 – steam generator; 3 – deaerator; 4 – turbine; 5 – electric generator; 6 – main condenser; 7 – low-pressure regenerative heaters; 8 – high-pressure regenerative heaters; 9 – thermal energy storage system; 10 – oil charge heat exchanger; 11 – high-pressure oil discharge heat exchangers; 12 – low-pressure oil discharge heat exchangers; 13 – oil storage tank; 14 – separator; 15 – overheater; 16 – pump; 17 – turbine stop-control valve.

So, the steam from the turbine regenerative bleed-off is partly used for heating of turbine main condensate and feed water or completely not used. And this steam can be used to increase the turbine power.

III. AUTOMATIC CONTROL SYSTEM

The proposed system which is shown on Fig. 2 consists of 2 regulators: the regulator of turbine power changing and regulator of feed water temperature into the steam generator inlet. On next page you can see the functional diagram of the proposed system.

This automatic regulation system of electrical current frequency in the power network works this way.

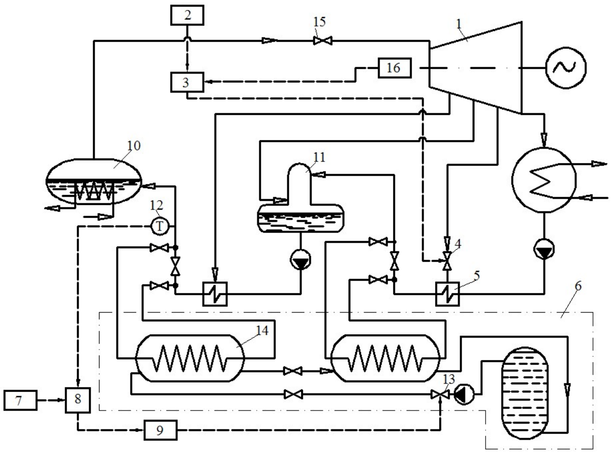

Figure 2. Functional diagram of the proposed automatic control system. In this diagram is indicated: 1– turbine; 2 – group current frequency network controller; 3 –turbine power changing regulator; 4 – control valve for changing turbine power; 5 – low-pressure regenerative heaters; 6 – thermal energy storage system; 7 – feed water temperature setting device; 8 – summator of preset and current feed water temperature; 9 – regulator of feed water temperature at the steam generator inlet; 10 – steam generator; 11 – deaerator; 12 – feed water temperature detector; 13 – oil control valve. 14 – oil heat exchanger; 15 – turbine stop-control valve; 16 – current turbo-generator power.

Changing the network power load leads to the changing of electrical current frequency in the power network. The group current frequency network controller (2) distributes the required power changing between all power units which participate in regulation of electrical current frequency in the power network. The signal of the required power change (ΔN) for our NPP in the discharging mode of the TESS (6) is transmitted to the turbine power changing controller (3), which changes the position of the control valve (4) located on the turbine regenerative bleed-off. This leads to changing of the steam flow rate in the turbine regenerative bleed-off, and then to changing of the steam flow rate through the turbine (1). And in the end this changes the turbine power. As the result proposed system ensures the participation of the NPP in the regulation of electrical current frequency in the power network with the required rate.

At the same time changing of steam flow rate to the low-pressure regenerative heater (5) leads to water temperature changing from the heater (to the deaerator inlet (11)). Changing of temperature is compensated in the deaerator, where a constant steam pressure is maintained. This means that we have the required feed water temperature at the deaerator outlet.

The summator (8) compares the current feed water temperature at the steam generator inlet (10) and the preset (nominal) temperature. In the case of a non-zero signal (ΔT) from the summator (8), the regulator of the feed water temperature (9) changes the position of the control valve (13) located on the oil pipeline of the TESS (6). This leads to the oil flow rate changing through the oil heat exchanger (14) and ensures the constant feed water temperature at the steam generator inlet (10).

Therefore, the change of the reactor power due to the movement of the control rods is eliminated (due to the maintenance of constant steam flow rate and steam pressure in front of the turbine) [3].

IV. CONCLUSIONS

The theme of the presentation is dedicated to the innovative automatic control system in the nuclear power industry, developed for existing NPP's with VVER under the modernization, and the future projects of NPP. This system will provide:

- achievement of the technical efficiency of the NPP units with TESS in comparison with current NPP units (without TESS),
- increase of power productivity by means of its additional participation in the regulation of electrical current frequency in the power network without changing of the reactor power,
- increase of reactor reliability (the reactor works at the constant power all the time) and hence the power unit safety in whole.

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Software and Hardware Complex “Virtual NPP” as an Instrument of Scientific and Technological Assistance to the Nuclear Operating Company

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I. Introduction

Currently, the state-of-the-art mathematical simulation methods and deep understanding of the physics of phenomena occurring in the equipment, as well as capabilities of modern computing systems allow to provide full-scope multiphysics simulation of processes occurring in the NPP power unit. The range of reproducible modes of the power unit may be very wide – from normal operation to severe accidents.

In order to develop mathematical models of power units that are capable, on the one hand, of high-fidelity simulation of physical processes in the widest range of possible operation modes, and, on the other hand, taking into account simultaneous operation of a large number of technological process systems and auxiliary systems, including a detailed I&C model, it is necessary to develop a modern system of calculation codes (or to adapt the existing ones), which are based on the possibility of parallel calculations on a distributed computing platform consisting of computing servers of different types (including super-computers).

An adequate modern system of calculation codes for simulation of various processes and phenomena occurring in the equipment of VVER power units, adapted for high-performance computing, was developed along with the hardware and software complex "Virtual NPP" (hereinafter – H&SC VNPP).

The closest analogue of H&SC VNPP is the system of calculation codes developed within the CASL project [1]. Unlike this system, H&SC VNPP is focused on simulation of severe accident processes, and provides full-scope simulation of the power unit with a detailed account of the operation of the turbine equipment, electrical equipment and automation systems.

For the assessment of qualitative and quantitative characteristics of the developed system of calculation codes within H&SC VNPP, a full-scope model of Novovoronezh NPP Unit 6 was developed. This model was used to reproduce a large number of modes occurring on a real power unit at the test stage, and for further comparison of the simulation results with the operational archives of the power unit. The comparison showed a high level of adequacy of the VNPP-based model, which allows describing various modes of power unit operation with high-fidelity.

II. Structure of the Hardware and Software Complex "Virtual NPP"

H&SC VNPP is a universal platform for the simulation of VVER-based power units. H&CS VNPP may serve as a basis to build mathematical models of power units, which allow multiphysics simulation the following processes and phenomena:

- neutronic processes in the core;
- thermohydraulic processes in the equipment of the reactor and turbine islands;
- electromechanical processes in the electric equipment of the power unit;
- operation of power unit I&C;
- processes associated with the accumulation and propagation of fission products in the power unit and throughout the adjacent territory in emergency conditions, including fuel rod cladding leakage, the behavior of fission products in the gas and aerosol phases in the coolant circuits and in the containment;
- processes occurring in the reactor plant during a severe accident (core destruction, melt behavior on the vessel bottom, corium melt-through, melt behavior in the core catcher or on the surface of the concrete shaft of the reactor).

Figure 1 shows the scheme of calculation codes included in the H&SC VNPP [2].

The calculation basis of the H&SC VNPP are Russian-made computer codes, which are further development of the calculation base currently used to build NPP unit simulators and to assess the safety of reactors.

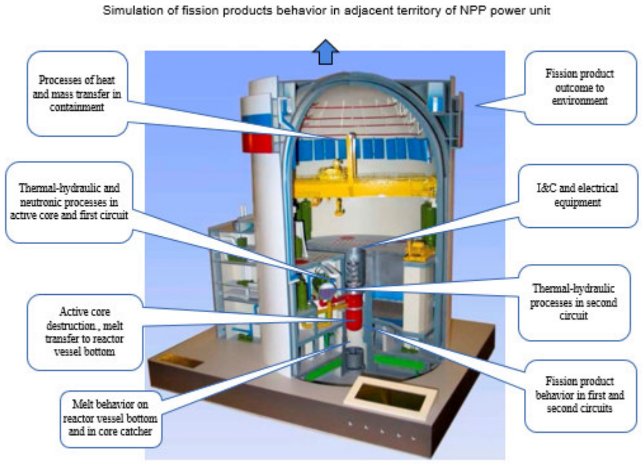

Fig.1. Scheme of calculation codes in H&S VNPP

H&SC VNPP contains calculation codes of different levels of complexity and speed:

- the high-speed level allows building full-scope power unit models that can perform real-time calculations or even faster;
- the high-fidelity level is based on the application of more detailed approximations compared to the high-speed level; this level considers a wider range of processes and phenomena covering the field of severe accident regimes;
- the precision level allows to provide simulation the state of individual equipment by using a detailed partition of the computational domain (up to a billion of individual control volumes).

On the basis of H&SC VNPP, it is possible to build integrated models that combine the capabilities of different levels of modeling.

Neutronic models used at the high-speed level and high-fidelity level are based on the application of the three-dimensional small-group diffusion approximation with the possibility of modeling spatial kinetics. At the precision level, the transport approximation solved by the Monte Carlo method is applied.

The thermal-hydraulic models used at the high-speed and high-precision level are based on the application of one-dimensional channel approximation.

At the high-precision level, thermal-hydraulic modeling of the core is possible with use of two-dimensional and three-dimensional approximations.

At the precision level, the CFD modeling approach is applied.

Electromechanical models allow to provide simulation the main equipment of the power unit as electricity consumer. It is possible to simulate short circuits on consumer sections and in separate cable lines, partial or full blackouts, disturbances in the external network.

The automation system is emulated at the level of application software, which is used in the automation equipment at a real power unit. This approach allows keeping track of the work of automation systems without any simplifications to their real logic of operation.

The fission products propagation processes are modeled in view of possible propagation of fission products in the gas and aerosol phases, and as impurities in the coolant. When modeling fission products in the aerosol phase, processes such as condensation, deposition, coagulation, and nucleation are taken into account.

Severe accident processes are modeled in a two-dimensional cylindrical approximation based on the solution of the heat and mass transfer problem [3]. When modeling a severe accident, the processes of fission products release from the melt, as well as the processes of hydrogen generation are taken into account. Radiation doses to the operation staff in various premises are estimated, and also the probability of detonation of the hydrogen gas mix within the containment is calculated.

All of the above processes are simulated in multiphysics way, taking into account the simultaneous operation of a large amount of equipment.

Seamless simulation of the emergency is possible, starting with the initial effects (failures) in normal operation up to the late stages of severe accidents.

VNPP-based mathematical models of power units may function on a distributed computing environment within H&SC VNPP, which can significantly reduce the time of simulation of complex processes.

An additional feature of H&SC VNPP is the ability to image the calculated information on many different visualization tools (see Fig.2), which not only facilitates the process of analyzing simulation results, but provides additional functionality (e.g., verification of the human-machine interface of the main control room).

III. Practical Application of the Hardware and Software Complex "Virtual NPP"

H&SC VNPP was developed for scientific and engineering support of nuclear operators, which implies solving the following important practical problems:

- computational verification of power unit designs (both new ones and those to be modernized); it is expected that changeover from the expert analysis of design documentation to analysis with use of mathematical models of power units will reduce the number of inconsistencies in the design at its early stages and, as a consequence, reduce the timing of commissioning and increase the level of efficiency and safety of industrial operation of power units.

- development of new-generation educational aids; improved fidelity and expanded range of reproducible modes covering severe accidents will improve the quality of training of operation staff, and increase the level of involvement of experts from emergency response centers into the educational and training process;
- design verification and optimization of emergency response instructions will confirm the adequacy of the developed instructions for power unit operation in emergency situations and, if necessary, make reasonable changes based on calculations, thus implementing better emergency management.

It is worth noting that H&SC VNPP may serve as the basis for the development of full-scope mathematical models of power units, which are essentially digital twins of power units. Such models may be of interest not only in the context of the problems listed above, but also may be considered as an independent supplement to the design documentation of the power unit.

At the time of this writing, underway are the initial stages of the pilot projects of H&SC VNPP application for the verification of design solutions in the I&C part of the power unit, as well as its application for the development of a mathematical model of the unit to be integrated in the full-scope simulator and analytical simulator of the prototype power unit.

IV. Conclusion

The hardware and software complex "Virtual VVER-based NPP" is a modern tool for full-scope simulation of the VVER-based power unit. Moreover, the composition of the model allows it to adequately reproduce a wide range of possible operating modes of a real power unit, from normal operation up to late stages of severe accidents.

H&SC VNPP is considered as an effective tool for scientific and engineering support to nuclear operators (Regenerator, JSC) and is coming into use for the development of educational aids for the operation staff of new-generation nuclear power units, as well as for the purposes of computational verification of power unit I&C designs.

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Fig.2. General view of the H&SC "Virtual VVER-based NPP"

Improving the safety and economic efficiency of Russia's nuclear power plants by the example of Balakovo NPP

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I. Introduction

Rosenergoatom is the operator of all nuclear power plants in Russia. Today it includes 37 operating power units with a total capacity of 30.1 GW. For the last 10 years, the concern has been pursuing a policy of increasing economic efficiency, while constantly enhancing safety.

Using the example of Balakovo NPP, I want to show how Rosenergoatom achieves the objectives of this policy.

Balakovo NPP is located in the city of Balakovo, about 900 kilometres south-east of Moscow. It consists of four power units with PWR called VVER-1000. Aggregate installed capacity 4000 MW. Commissioning date: power unit No.1 - 1985, unit No.2 - 1987, unit No.3 - 1988, unit No.4 – 1993.

II. Increasing power generation and improving economic efficiency

A. Service life extension

Service life extension for existing NPP units is one of Rosenergoatom's priorities aimed to maintain generating capacities and increase NPP safety. As of October 1, 2019, there were 26 NPP units with extended service life under operation with the total installed capacity of 17.8 GW [1]. At the Balakovo NPP, 3 out of 4 units are operated with an extended service life. This was made possible by assess safety, technical capability, integrated survey of the systems, elements and equipments resource characteristics. As result was implemented the program to prepare the units for extra service life, involving large-scale upgrading and improvement of safety level in accordance with up-to-date global standards and recommendations of the IAEA.

<u>Highlights of the program</u>

1) *High temperature annealing of a reactor vessel on power unit No. 1.* The long-term operation of PW reactors under neutron irradiation leads to radiation embrittlement of ferritic pressure vessel steels. This limits the lifetime of the whole NPP. A thermal anneal cycle above the normal operating temperature of the vessel can recover the structure and mechanical properties of the pressure vessel metal to its original state. The metal in the reactor vessel was slow heated to a temperature of +565 degrees Celsius, after which it began a stationary annealing, which lasted 100 hours. Then, the metal was slowly cooled to a temperature of +25 degrees Celsius [2]. According to the annealing results, reactor vessel on power unit No. 1 received an additional 15 years of service life.

2) *Upgrade of main reactor, turbine, electrical, measurement and control equipment, process systems, control and protection systems, air conditioning and ventilation systems.*

3) *Replacement of NPP equipment that reached the end of its service life with a view to increase reliability.* For instance the largest activities are :

- Replacement of turbine condensers - the biggest heat exchangers at NPP (dimension: 20.5 meters length, 16.7 width meters, 10.5 height meters.) For this purpose, the turbine building wall panels were dismantled [3] (Fig.1). The condensers with heat exchange tubes from copper-nickel alloy were replaced condensers with corrosion resistant steel tubes for power unit No.1. and with titanium alloy plated carbon steel tubes for power units No 2,3,4.
- Replacement of the generator stator (weigh 333 tons) for power units No 2,4.

Figure 1. Dismantled wall panels of the turbine building

B. Increasing of thermal power and transition to longer fuel cycles.

Another way to improve economic efficiency of NPP is increase of thermal power and transition to longer fuel cycles. One of the options to increase power generation at NPP power units with VVER-1000 reactors consists in increasing heat capacity of reactor plants using engineering reserves of equipment taking into account the actual characteristics obtained upon its manufacturing and recorded during operation. Initially the VVER-1000 reactor project was designed for heat capacity of 3,250 MW. However, the previously developed design of the generator had an electric capacity of 1000 MW. That is why based on power efficiency of the entire turbo-generator unit the heat capacity of VVER-1000 reactor was limited to 3,000 MW. Increase in heat capacity is implemented due to intensification of fuel fission. In this case, the temperature in the reactor has increased by 1.5 degrees Celsius — from 320 to 321.5 degrees Celsius in average in the active zone [4]. By increasing the temperature of water at the output of the reactor it is possible to submit more heat to the steam generator. In this case the parameters inside the steam generator (pressure and temperature) remain the same. The volume of generated steam increases. In order to ensure that more volume of steam is produced, the turbine wheel space and last buckets were also modernized. Also, deeper dehydration of resulting steam from steam generators is provided for. The increased capacity of generator, while the terminal voltage remains the same, is a consequence of heavy currents. In order to cool down the ironed generator due to heat released under higher current load, the density of the cooling gas (hydrogen) was increased. During operation of power units at an increased capacity conducted and continue conducting an analysis of the main parameters of reactor unit to ensure compliance with permissible operating limits. According to this analysis, the values of neutronic and thermohydraulic characteristics of the active zone are compliant with design parameters and meet the requirements and design limitations. Today all 4 power units Balakovo NPP operated at an increased (104%) capacity level.

In order to implement the cost-effective longer fuel cycles, the application of new types of fuel assemblies with higher 4.90-4.95% enrichment in Uranium-235 and the enlarged fuel column by 150 mm were required [5]. As a result, the 12-month fuel cycle increased to the 18-month fuel cycle.

All upgrades described above led to raised up power generation

III. POST FUKUSHIMA SAFETY ENHANCEMENT

Rosenergoatom always declares that safety is a top priority. The Fukushima accident once again showed that the main task is to ensure 3 fundamental safety functions:

- Reactivity control (prevention of uncontrolled power increase and ensuring safe emergency shutdown of the reactor if necessary).
- Removal of residual heat (shutdown reactor cooling and spent fuel store cooling).
- Confinement of radioactive material (shielding against radiation and control of planned radioactive releases, as well as limitation of accidental radioactive releases).

In connection with the events at Fukushima NPP, Rosenergoatom implements activities to ensure protection of nuclear plants against extreme external impacts. As the most important and large-scale activities among efforts to mitigate consequences of hypothetical, unforeseen emergencies on Balakovo NPP, were equipped with mobile anti-emergency equipment. Nuclear power plant received mobile diesel generator units (MDGU) used in standby power supply for consumers of systems important to safety (spent fuel pump, emergency boron injection pump), monitoring, control and protection systems, emergency lighting and communication of power units during beyond design basis accidents. Also, Balakovo NPP received mobile pumping units (MPU) and monoblock pumps.

- MPU 150/90 for boric acid solution supply and cooling primary circuit.
- MPU 40/50 for boric acid solution supply and cooling spent fuel pool.
- MPU 150/120 for water supply and cooling steam generators.
- MPU 500/50 for water supply to essential service water system
- Monoblock pumps use for pumping underground level of reactor, turbine and emergency diesel generator buildings.

All these equipment are used for beyond design basis accidents when normal power supply lost and emergency diesel generator fails.

Figure 2. Mobile diesel generator units 2 MW

Figure 3. Balakovo NPP Performance indicators

IV. Results of Safety and Economic Efficiency Improvement Policy

Rosenergoatom`s safety and economic efficiency improvement policy leds to impressive results. As of January 1, 2019 942.28 bln KWh of electricity produced by power units with the extended service life, 27.62 bln KWh –additional energy generation on reactors operated at an increased capacity. Increase of power capacity by 1 kW is approximately 10 times cheaper than the construction cost of 1 kW of new power capacity [1].

Cost-efficient power generation while assuring the safety priority - main goal of Rosenergoatom activity. During the period from 1995 to 2019 there are only 5 events with the level "1" at the INES scale On Blalkovo NPP, other deviations in operation of NPP are at the level "0" or "below scale" at the INES scale. Fig. 3 shows that all the improvement of the economic efficiency on Balakovo NPP, described above, were not affected in any way by the increase in the number of failures and incidents. On the contrary, there is a tendency to reduce their number.

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Nuclear Reactor Operator Pregnancy: Managing Risk While Maintaining High Functional Capacity

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I. INTRODUCTION

The safe operation of the Open Pool Australian Light-water Reactor (OPAL) supports the needs of Australian and international communities through the provision of neutrons for human health, science, and industry. Radiopharmaceuticals originating from the OPAL Reactor are used for the diagnosis and treatment of disease; scientists drive innovation through neutron scattering experiments reliant on access to neutron beam lines; and silicon ingots are irradiated in the OPAL Reactor for industrial suppliers to distribute for use in the manufacture of high quality semiconductor devices.

Several hazards typical in a nuclear facility–including radiation, electrical hazards, work at heights, manual handling, chemicals, cryogenics, biological hazards, and fatigue–are present at the OPAL Reactor. In order to maintain a high level of safety assurance, area- and task-based hazard analyses and risk assessments are performed to identify risks to workers and other persons, and to specify associated controls. In the case of a pregnant worker, the risk management process must consider not only the worker but also the unborn child or children.

The aim of this paper is to describe a flexible, risk-based approach that was used in managing risk to OPAL's first pregnant operator. The approach used at OPAL is contrasted with what may be considered a typical way of managing risk to a pregnant radiation worker. The paper includes some comments regarding perception management in the context of a work team that includes a pregnant radiation worker.

II. OPAL REACTOR OPERATIONS ROLES

The OPAL Reactor Operations section is responsible for the safe and reliable operation of the OPAL Reactor. The functional roles within the section are those of Reactor Operator, Reactor Engineer, Shift Manager, and Reactor Chemist.

Reactor Operator responsibilities include: operation of all reactor-based plant and equipment from the Main Control Room or in the field; performance of electrical and process isolations; release of plant for maintenance; monitoring of all activities within the facility; performance of minor maintenance activities; performance of radiological protection tasks; response to all abnormal conditions including emergencies; application of technical and diagnostic skills to improve reactor safety and reliability; and performance of complex investigations.

While the Reactor Operator and Reactor Engineer roles share a common accreditation and authorisation program, the roles differ in that the Reactor Engineer role requires a minimum professional qualification in the form of a bachelor degree in engineering or science, and it exists as a development role; Reactor Engineers with demonstrable technical and leadership competencies may qualify to progress to the Shift Manager role, with the process typically taking approximately five years. During this time the Reactor Engineer performs the duties of a Reactor Operator and has additional responsibilities.

III. MANAGEMENT OF RISK TO PREGNANT WORKERS

A. A Generic Approach and its Restrictions

One of the primary objectives of the Australian Work Health and Safety Act 2011 is to "secure the health and safety of workers and workplaces by protecting workers and other persons against harm to their health, safety and welfare through the elimination or minimisation of risks arising from work" [1]. In the nuclear industry where radiological and other industrial hazards are present, management of risk to the pregnant worker and the unborn child has often been implemented by elimination of the hazard, with this being achieved through the temporary removal of the pregnant worker from their hazard-exposed function. While the hierarchy of controls states that elimination is the most effective means of controlling a hazard, the approach of removing the pregnant worker from their hazard-exposed function may not be the optimal way of managing risk in this context. Such a generic approach has the potential to be overly conservative, as it fails to take into account the specific circumstances of the worker and the particular hazards of the work environment. Importantly, this approach fails to involve the worker in a decision-making process regarding potentially significant career path changes.

B. ANSTO Guidance for Pregnant Workers

The ANSTO guidance aims to protect the health and safety of the worker and of the unborn child in alignment with the WHS Act [1]. This section outlines key points contained in the ANSTO guidance. Importantly, the ANSTO guidance is not unreasonably prescriptive in terms of specific measures to be implemented upon disclosure of a worker's pregnancy. Rather, the guidance directs the pregnant worker and the relevant manager(s) and advisors to appropriately assess the pregnant worker's specific situation and implement suitable controls for identified hazards.

ANSTO Guide AG-5598 [2] details the recommended disclosure process for any pregnant worker at ANSTO. The guide encourages the pregnant worker to advise their supervisor or manager as soon as reasonably practicable so that appropriate measures can be taken to control potential exposure to work-related risks. AG-5598 states that a risk assessment must be conducted on all activities that the pregnant worker will perform, and that the risk assessment is required to consider the working environment as well as the use of physical, chemical, biological, and radiological agents.

ANSTO Procedure AE-2310 [3] states that pregnancy or breastfeeding shall not be considered a reason to exclude the worker from work. It also states that working conditions with regard to occupational exposure shall be adapted "to ensure that the unborn child or the breastfed infant is afforded the same broad level of protection as is required for members of the public" [3]. The annual effective dose to a member of the public must not exceed 1 millisievert, with the ANSTO public exposure annual effective dose constraint being 0.75 millisieverts [3]. AE-2310 further states that "ANSTO is committed to keeping the likelihood of incurring exposures, the number of people exposed, and the magnitude of their individual doses as low as reasonably achievable, taking into account economic and societal factors (the ALARA principle, ICRP 103 (2007))" [3], [4].

ANSTO Guide AG-2677 [5], provides further information and clarification on the topic. However, it is noted that AG-2677 was not available at the time of the pregnancy discussed in Section IV. AG-2677 reiterates the statements that neither pregnancy nor breastfeeding shall be considered a reason to exclude the worker from work, and that being pregnant does not necessarily mean that the worker must avoid all work with radiation or radioactive materials. AG-2677 includes examples of precautions that may be taken when a pregnant worker is working with radiation, including temporary reassignment to tasks in areas with lower dose rates, and use of an electronic dosimeter with a dose rate alarm.

IV. OPAL'S FIRST PREGNANT REACTOR OPERATOR

A. Disclosure of the Pregnancy

In 2017, an OPAL Reactor Engineer disclosed her pregnancy to the OPAL Operations Manager in accordance with the ANSTO guidance, with the disclosure occurring early in the first trimester. Because early disclosure of a pregnancy can be an extremely sensitive matter, subsequent disclosure to additional staff members occurred in stages, as detailed below, in order to maintain confidentiality of the pregnancy as far as reasonably practical.

In the interests of optimising protection of the health and safety of the Reactor Engineer and her unborn child, the Reactor Engineer and her manager agreed to disclose the pregnancy to two ANSTO employees: the Work Health and Safety (WHS) senior manager and the Radiation Protection Services (RPS) senior manager. This further disclosure, which occurred almost immediately after the initial disclosure to the Operations Manager, enabled the Reactor Engineer and Operations Manager to develop comprehensive risk assessments that took into account expert advice. Informing these two senior section managers was considered preferable to informing local WHS and radiation protection staff because the senior section managers were somewhat removed from the immediate workplace, did not personally know the pregnant worker, and had a senior level of authority and responsibility in their fields to give advice and assist with review of risk assessments.

During the early stages of the pregnancy, while the Reactor Engineer was working on shift (see *B. Work Arrangements During the Pregnancy*), it became necessary for her to disclose her pregnancy to the duty Shift Manager to allow for optimised management of hazards during routine shift work.

At a point during the pregnancy when the Reactor Engineer was comfortable with the news being publically disclosed, subsequent consultation was held with ANSTO Health Centre staff, the OPAL Reactor Manager, Executive Officer to the CEO, and directly affected shift staff.

B. Work Arrangements During the Pregnancy

The Operations Manager and Reactor Engineer conducted an initial assessment of existing and potential work arrangements. The assessment considered: the health and safety of the Reactor Engineer and her unborn child; the requirement to keep news of the pregnancy confidential during the early stages; the Reactor Engineer's desire for ongoing learning and development opportunities; the fact that the Reactor Engineer was, at the time, working on shift as part of minimum staffing; and the fact that any decisions and arrangements may need to be reviewed as the pregnancy progressed.

The Operations Manager presented the option of allowing the Reactor Engineer to temporarily discontinue shift work with the proposed public justification being the requirement to work on an urgent, special project. However, the Reactor Engineer's preference was to remain on shift for the time being, as this would allow her to continue on her intended career path until any changes became necessary or preferable as a result of the pregnancy.

Upon agreement that the Reactor Engineer would continue working on shift, the Operations Manager and Reactor Engineer promptly discussed and drafted risk assessments encompassing both the radiological aspects and the WHS aspects of ongoing shift work during pregnancy. The risk assessments were reviewed in consultation with the RPS senior manager and WHS senior manager respectively. Appendix 1 presents a summary of key points from these risk assessments. It was acknowledged that the risk assessments may require review to appropriately accommodate changing circumstances throughout the pregnancy. Furthermore, it was agreed that, if requested by the Reactor Engineer, ANSTO would accommodate any change in work arrangements required to remove the Reactor Engineer from industrial or radiological hazards, and that other suitable tasks would be assigned.

The Reactor Engineer continued working on shift until just after midway through the pregnancy, at which time she elected to discontinue shift work and perform duties better suited to her current circumstances. The Operations Manager was supportive of this decision, and alternative work arrangements were made. These alternative arrangements included working standard work hours and, in general, not working in potentially hazardous work environments.

C. Managing Perceptions During the Pregnancy

When the pregnancy was publically disclosed and while the Reactor Engineer continued working in her role on shift, there was a very high level of supportiveness and positivity received from colleagues both in the immediate work team and across the division.

However, whether as a result of curiosity, concern, lack of awareness, established opinions based on prior experiences and observations, or other reasons, some staff members raised questions (either directly to the Reactor Engineer or to the Operations Manager) regarding the appropriateness of the pregnant Reactor Engineer continuing to work on shift and perform routine reactor operation tasks. Staff members trained in risk management of workplace hazards, industrial hazards, and radiological hazards were generally reassured upon becoming aware of the nature and depth of the risk assessments that had been developed. Nevertheless, despite these comprehensive risk assessments and the control measures in place, there remained some staff members who continued to question the appropriateness of the Reactor Engineer's ongoing work in hazardous environments.

V. CONCLUSION

With regard to pregnant workers in work environments that contain radiological hazards and industrial hazards, it has been demonstrated that the key objective of the WHS Act [1] can be achieved through a flexible approach that does not require full and immediate removal of the pregnant worker from their hazard-exposed function. Through consultation and the use of appropriately comprehensive risk assessments, the demonstrated approach thoroughly manages risk to the pregnant worker and to the unborn child whilst supporting the worker's right to maintain a level of control and influence over their ongoing career development. It is likely that opinions will remain divided with regard to the appropriateness of a pregnant worker performing work in hazardous environments. However, the increasing gender diversification within traditionally male-dominated workforces may in time result in a wider acceptance of flexible approaches to managing workplace risks during pregnancy.

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[5] ANSTO Guide AG-2677, "Working safely with ionising radiation – guidelines for expectant and breastfeeding mothers".

[6] OPAL Manual OM 06, "Radiation protection plan"

[7] OPAL Instruction OI 21, "Requirements for personnel entering or exiting OPAL radiologically classified areas"

[8] OPAL Operations Manual OOM 0057-001, "Radiological classification and identified safety hazards for the OPAL reactor"

[9] OPAL Instruction OI 13, "Access, working and exiting forbidden and restricted areas"

[10] OPAL Instruction OI 26, "Electrical work, switching, and isolation"

APPENDIX 1 – RISK ASSESSMENT SUMMARY

Table 1 presents key items from the risk assessments referred to in Section IV Part B. Not all items covered in the risk assessments have been included here. Risk ratings have been omitted since these are meaningful only in the context of the relevant risk matrix and associated guidance for specification of risk ratings.

TABLE I. RISK ASSESSMENT SUMMARY

Activity	Hazards	Controls
Travel to and from work as a shift worker	Potentially significantly affected general fatigue levels and sleep patterns Potentially increased risk of car accident	• Maintain awareness of potentially increased fatigue levels during pregnancy. • Avoid driving or stop driving if feeling fatigued when travelling to or from work. • Use General Leave as required. • Reinforce current controls already implemented and trained, including: use of Cabcharge cards; healthy lifestyle; sleeping aids; rest breaks; and management of work.
Work undertaken in radiological areas	(In addition to hazards already documented and controlled) Exposure to unborn child	• Continue to work in accordance with: OM 06 – Radiation protection plan [6]; OI 21 – Requirements for personnel entering or exiting OPAL radiologically classified areas [7]; OOM 0057-001 – Radiological classification and identified safety hazards for the OPAL reactor [8]; OI 13 – Access, working and exiting forbidden and restricted areas [9]. • Perform activities involving only *routine* access to radiologically classified areas. • Avoid activities that may result in a non-typical radiological daily exposure. (Examples were specified in the relevant risk assessment.) Share tasks with other staff if daily dose is likely to exceed typical gamma effective dose (as specified in the risk assessment). • Use a personal portable dose rate monitor when practicable. Wear an Electronic Personal Dosimeter (EPD) and Thermoluminescent Dosimeter (TLD) at all times. • Carefully manage access to areas containing potentially tritiated heavy water and avoid exposure to airborne tritiated heavy water. • Avoid routine work requiring Respiratory Protective Equipment (RPE). If Self-Contained Breathing Apparatus (SCBA) is required during emergency response, alternative staff should be used. The only RPE that should be worn is the Escape Emergency Respiratory Device if required. • Avoid activities requiring the use of Tychem or Tyvek suits or similar. • Maintain a record of daily EPD readings. Inform Operations Manager at 100-microsievert intervals, if reached. Operations Manager will consult with RPS senior manager if 350 microsieverts (total) is reached. Dose will be kept as low as reasonably achievable and ANSTO dose constraint of 750 microsieverts will not be exceeded.
Serious event or emergency response	Psychological (stress) Radiological Physical (injury) Fire Chemical Cryogenic Electrical	• It is recommended that the pregnant worker not be exposed to any hazard from a serious escalating event or from any emergency. • If the worker desires, she may provide support from the Main Control Room if the situation is unlikely to result in significant stress. It is recommended that additional support be contacted. • The worker may inform the duty Shift Manager if she opts to be excluded from emergency response activities. The Shift Manager will arrange for additional support.
Electrical switching	Arc flash Electrocution Potential change in risk of human error due to increased fatigue	• Continue to work in accordance with OI 26 – Electrical work, switching, and isolation [10]. • Continue to use existing engineered controls. • Maintain awareness of potentially increased fatigue levels. • Take more frequent breaks.
Working at heights	Potentially increased risk of fall due to fatigue Potentially increased risk to unborn child if fall occurs with harness	• Do not perform any work that requires use of harness ladders. • As far as practicable, avoid work that requires the use of step ladders.
Working with or near chemicals	Unanticipated exposure to chemicals	• Avoid any work with chemicals where exposure or a spill could be harmful. • Apply a higher level of discretion in areas with significant quantities of chemicals.
Work involving manual handling / reaching / stretching	Additional risk of strain or harm to pregnancy	• Avoid any significant manual handling activities such as gas cylinder movements or lifting of liquid nitrogen dewars. • Request other staff complete such tasks or use suitable aid if appropriate.
All work activities	Potential for psychological stress, particularly due to the need to maintain confidentiality of the pregnancy in the early stages	• Inform the Operations Manager if any issue arises or if any support is required. • Consider obtaining support from the Employee Assistance Program and ANSTO Health Centre if required. • Operations Manager: Provide ongoing monitoring and consultation.

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Method for calculating the thermal power of the "PIK" reactor

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I. INTRODUCTION

The high neutron flux research reactor "PIK" is designed for a wide range of tasks in the field of nuclear physics and weak interaction and condensed matter, structural and radiation biology and biophysics, radiation physics and chemistry, as well as for solving applied technical problems. "PIK" is light water-cooled reactor with central neutron trap placed inside the reactor's core which is surrounded by heavy-water moderator. The main reactor parameters are given in the table I.

TABLE I. MAIN PARAMETERS OF THE "PIK" REACTOR

Parameter	Value
Maximum thermal power, MW	100
Volume of the core, m^3	0,05
Maximum neutron flux, neutron/cm^2·s	$4{,}5\cdot10^{15}$
Neutron flux in the moderator (reflector), neutron/cm^2·s	$1{,}3\cdot10^{15}$
Pressure, MPa: – primary curciut (H_2O) – intermediat curciut (H_2O) – third curcuit (H_2O) – moderator (D_2O)	 5,00 0,27 0,57 0,30
Fuel enrichment, % U^{235}	90
Horizont experiment channel, units	10
Slope experiment channel, units	6
Vertical experiment channel, units	6
Cold neutron source, units	1
Fast neutron source, units	1
Neutron guide, units	10
Neutron flux at the neutron guide output, units	$(1{,}1–1{,}4)\ 10^{10}$
Amount of research equipment which can be installed at the same time, units	50

When mastering the power of the "PIK" reactor over 10 MW during commissioning, it is necessary to calibrate NFME (Neutron Flux Monitoring Equipment) channels.

NFME calibration is basically an adjustment of each monitoring channel by calculated power value, and one of the well-known method to calculate reactor's thermal power is indirect method through measured thermohydraulic parameters.

The gamma-spectrometric method, which is currently used for determining the reactor power, is applicable only for the power range up to 100 kW, as it is radiation hazardous, and it requires depressurization of the core to extract irradiated samples, which technically does not provide real-time parameters control. Furthermore, it is prohibited to use the existing compound fuel assemblies for gamma-spectrometric method during coolant circulation in the core.

On industrial power reactors of the VVER type, control of the thermal power of the reactor and the thermohydraulic parameters of the primary circuit in the range from 10 to 120 % N_{nom} of the nuclear reactor power is the task of the IRMS (Internal Reactor Monitoring System). There are not any analogues of the IRMS in the «PIK» reactor because it's core is too small and doesn't let an installation of additional equipment there, so the parameter «thermal power» is not available for reactor operator now.

To understand how much energy reactor's core produces, we suggest to summarize the amounts of the thermal energy which is removed from core to heat-exchangers by every cooling system

II. ENGINEERING FLOW DIAGRAM

The main equipment of the «PIK» reactor and it's orientation are showed at the fig. 1. This sketch shows the main systems, which are thermal energy transferrers:

1 – Central Experimental Channel (CEC), which is placed in the center of the core;

2 – Primary Circle, which consists of three cooling circulation legs;

3 – Heavy Water Moderator Tank (HWM), which surrounds the reactor's core;

4 – Metal-Water Protection (MWP), which flows over the HWM;

5 – Liquid Regulator (LR), which is placed between HWM and reactor's core.

All the systems have their own heat-exchangers.

The scheme of the primary circle's pipes and main thermohydraulic parameters detectors location are showed at the fig. 2.

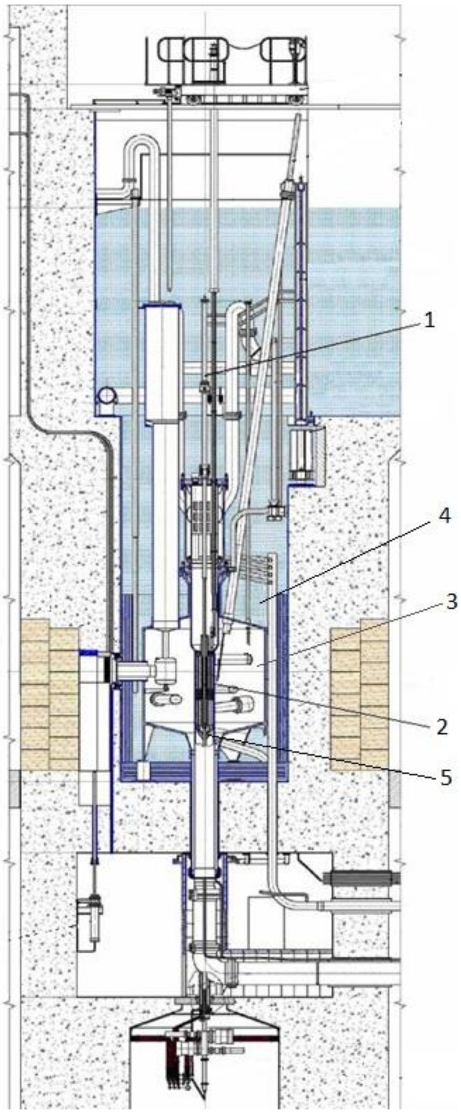

Figure 1. The «PIK» reactor sketch
1 – Central Experimental Channel; 2 – Core (Primary Circuit);
3 – Heavy Water Moderator Tank; 4 – Metal-Water Protection Volume;
5 – Liquid Regulator Volume

The temperature, pressure and flow detectors location allows to measure coolant heat-up through reactor’s core, and subsequently, to calculate the amount of thermal energy, which was transferred to coolant from nuclear fuel. For others cooling systems this method is also correct, but the difference is the thermal energy amount could be calculated through heat-exchangers instead of reactor’s core.

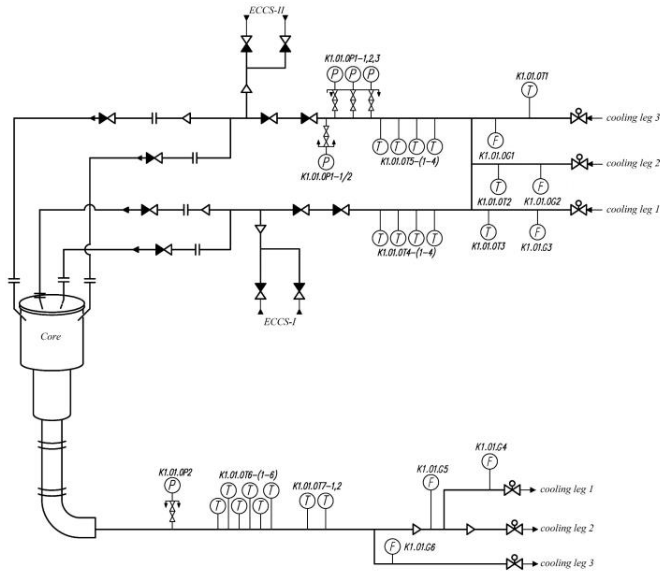

Figure 2. The primary circuit of the «PIK» reactor sketch:
ECCS-I, II – emergency core cooling systems I, II;
K1.01.0T1(2, 3) – temperature detectors on core input for cooling legs 1, 2, 3;
K1.01.0T4(5)-1–4 – temperature detectors on core input for upper/lower pressure header;
K1.01.0T6-(1–6), K1.01.0T7-1, 2 – temperature detectors on core output;
K1.01.0P1-1(2, 3), K1.01.0P1-1/2 – pressure detectors on core input (upper pressure header);
K1.01.0P2 – pressure detector on core output;
K1.01.0G1, 2, 3 – coolant flow detector on core input for cooling legs 1, 2, 3;
K1.01.0G4, 5, 6 – coolant flow detector on core output for cooling legs 1, 2, 3

III. Calculating Algorithm

Thermal power of the “PIK” reactor may be calculated with formula 1:

$$N_R = \sum_{i=1}^{3} N_i + N_{CEC} + N_{HWM} + N_{LR} + N_{MWP} + N_{TL} \quad (1)$$

N_i – thermal power transferred by cooling legs 1, 2 and 3, MW;

N_{CEC} – thermal power transferred by Central Experiment Channel (CEC) cooling system, MW;

N_{HWM} – thermal power transferred by Heavy Water Moderator (HWM) cooling system, MW;

N_{LR} – thermal power transferred by Liquid Regulator (LR) cooling system, MW;

N_{MWP} – thermal power transferred by Metal-Water Protection (MWP) cooling system, MW;

N_{TL} – thermal losses on equipment in condition of functioning ventilation system, MW.

In turns, N_i may be calculated with formula 2:

$$N_i = G_i^{mass} \cdot \left(h_{core}^{out} - h_i^{in}\right) \cdot \frac{1}{3,6 \cdot 10^6}, \quad (2)$$

$$G_i^{mass} = \frac{G_i^{Vin} \cdot \rho_i^{in} + G_i^{Vout} \cdot \rho_{core}^{out}}{2} \quad (3)$$

G_i^{mass} – mass flow through cooling leg-*i*, kg/h;

$G_i^{Vin,\,out}$ – volume flow at the input and output of cooling legs to reactor, m^3/h;

$\rho_i^{in}, \rho_{core}^{out}$ – water density at the input and output to reactor's core, kg/m^3;

h_i^{in}, h_{core}^{out} – water enthalpy at the input and output to reactor's core, kJ/kg;

$h_i^{in}, \rho_i^{in} = f(\overline{T_i^{in}}, \overline{P_i^{in}}),\ h_{core}^{out}, \rho_{core}^{out} = f(\overline{T_{core}^{out}}, \overline{P_{core}^{out}}),$

T_i^{input} – coolant temperature of each cooling leg at the input to reactor's core, ℃;

$\overline{T_{core}^{out}} = \frac{1}{3} \cdot \sum_1^3 T_{core-i}^{out}$ – arithmetical average of the temperature control channels at the output to reactor's core, ℃;

$P_{core}^{in} = \sum_1^3 P_i^{in}$ – coolant pressure at the input to reactor's core, MPa;

P_{core}^{out} – coolant pressure at the output to reactor's core, MPa.

The factor $\frac{1}{3,6 \cdot 10^6}$ is used here to get "MW" unit as a result.

The values of $h_i^{in}, \rho_i^{in}, h_{core}^{out}, \rho_{core}^{out}$ may be identified by WaterSteamPro code [3] or other ones with the similar tools.

Equivalent of the formula 1, thermal power transferred by others cooling systems may be calculated with formulas 4:

$$\begin{aligned}
N_{CEC} &= \frac{G_{CEC}^{V} \cdot \left(\rho_{CEC}^{in} + \rho_{CEC}^{out}\right)}{2} \cdot \left(h_{CEC}^{out} - h_{CEC}^{in}\right) \cdot \frac{1}{3,6 \cdot 10^6}, \\
N_{HWM} &= \frac{G_{HWM}^{V} \cdot \left(\rho_{HWM}^{in} + \rho_{HWM}^{out}\right)}{2} \cdot \left(h_{HWM}^{out} - h_{HWM}^{in}\right) \cdot \frac{1}{3,6 \cdot 10^6}, \\
N_{LR} &= \frac{G_{LR}^{V} \cdot \left(\rho_{LR}^{in} + \rho_{LR}^{out}\right)}{2} \cdot \left(h_{LR}^{out} - h_{LR}^{in}\right) \cdot \frac{1}{3,6 \cdot 10^6}, \\
N_{MWP} &= \frac{G_{MWP}^{V} \cdot \left(\rho_{MWP}^{in} + \rho_{MWP}^{out}\right)}{2} \cdot \left(h_{MWP}^{out} - h_{MWP}^{in}\right) \cdot \frac{1}{3,6 \cdot 10^6},
\end{aligned} \quad (4)$$

$G_{CEC,\,HWM,\,LR,\,MWP}^{V}$ – volume flow through CEC, HWM, LR and MWP cooling systems, m^3/h.

The water enthalpy and density of cooling systems are determined as like in the case of primary circuit:

$$\begin{aligned}
&h_{CEC,\,HWM,\,LR,\,MWP}^{in}, \rho_{CEC,\,HWM,\,LR,\,MWP}^{in} = \\
&f(\overline{T_{CEC,\,HWM,\,LR,\,MWP}^{in}}, \overline{P_{CEC,\,HWM,\,LR,\,MWP}^{in}}), \\
&h_{CEC,\,HWM,\,LR,\,MWP}^{out}, \rho_{CEC,\,HWM,\,LR,\,MWP}^{out} = \\
&f(\overline{T_{CEC,\,HWM,\,LR,\,MWP}^{out}}, \overline{P_{CEC,\,HWM,\,LR,\,MWP}^{out}}).
\end{aligned}$$

To simplify the calculation it's decided to admit N_{TL} as a constant. Quantity of N_{TL} will be specified during commissioning tests for each cooling system. At the initial approximation, the heat emission from primary circuit pipes in the hall is not more than $2,606 \cdot 10^4$ W.

Calculation errors evaluated according to papers [1], [2] as the errors of indirect methods could be estimated using formula 5:

$$\varepsilon(y) = \sqrt{\sum_j^n \left(\frac{\partial y}{\partial x_j} \cdot \varepsilon(x_j)\right)^2}, \quad (5)$$

$y = f(x_1, K, x_n)$ – function of *n* variables;

$\frac{\partial y}{\partial x_j}$ – partial derivative function y by j – variable;

$\varepsilon(x_j)$ – absolute uncertainty of j – variable.

Value of thermal power calculation errors depends on reactor power level and they may be inadmissible large on the low energy levels. Comparison of several independent methods, such as by gamma-spectrometric and coolant characteristics suggested algorithm will be informative in this case.

IV. Conclusion

Suggested algorithm provides to calculate thermal power of the research reactor «PIK» in full power range (with different calculating errors) in condition of coolant circulation through reactor's core and the other cooling systems.

During the following steps of "PIK" commissioning it is planned to conduct tests to provide validation of proposed method.

To provide the opportunity to calculate the "thermal power" parameter in real-time mode for reactor operators information support, it is necessary to integrate proposed algorithm to "DACCRS" (distributed automated complex of controlling reactor systems).

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KBA-KBC-1 System Upgrade for the Leningradskaya NPP-2

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I. INTRODUCTION

Nuclear energy development nowadays sets a mission to resolve vast variety of scientific and engineering tasks. One of the most important and difficult issues for today is nuclear power plant capability to follow the power system daily load curve (maneuver mode) with a wide range of power output variation from 40% to 100% $N_{nominal}$. In turn, this demands safety, effectiveness and economy justification of load following mode [1]-[3].

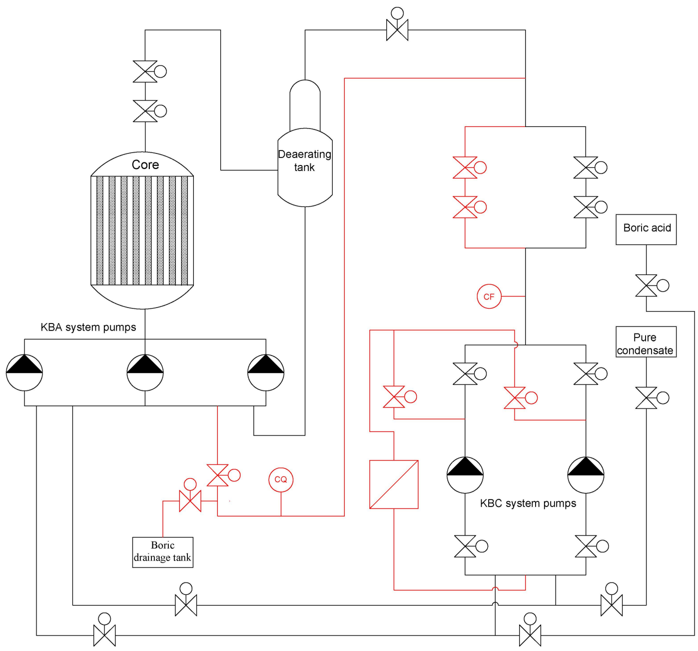
To justify the opportunity of nuclear power plant units with VVER-1200[1] reactor to be operated in daily maneuver modes, the test of four daily cycles with 100-80-100 % $N_{nominal}$ and one daily cycle with 100-50-100 % $N_{nominal}$ load curve were conducted. Test results show that NPP units with VVER-1200 are generally able to be operated in such modes, but under condition of feed&bleed and boron control system (KBA) and pure condensate system (KBC) upgrade. The KBA-KBC-1 systems upgrade is necessary for achieving following benefits:

– reduction in transport time (boric acid and pure condensate delivery from tanks to reactor core) from 40 down to 7-10 minutes;
– possibility to feed the boric acid and pure condensate with low flow rate and smooth flow rate change.

Coupled engineering flow diagram upgrade and automation level change at the same time make it possible to reach these aims.

The engineering flow diagram upgraded is shown at the fig. 1. As you can see, the alternate equipment is anticipated (red lines) in order to reduce boric acid and pure condensate transport time:

– motor-operated valves (low pressure);
– motor-operated regulation valves (low pressure);
– manual regulation valve (low pressure);
– safety plug (low pressure);
– back flow preventers (low pressure);



– plate-type heat-exchanger (low pressure);
– C&I (as in the text – Control and Instrumentation).

II. FUNCTIONAL GROUP CONTROL DESIGN

To provide automatic operation of upgraded KBA-KBC-1 systems the control algorithm was developed. This algorithm features a sequence of steps that combine into a structure called “Functional Group Control” (as in the text – FGC). FGC allows to make a centralized, effective (but not overloaded with numerous algorithms) way of KBA-KBC-1 equipment control, as well as it can be easily integrated into existing nuclear power plants control system. This algorithm makes possible to automate the equipment operation in greater way and to avoid the hard-to-trace faults occurrence in such manner that only supervision for normal systems running is required from reactor operator.

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[1] Pressurized water reactor with electrical power $N_{nominal}$=1200 MW

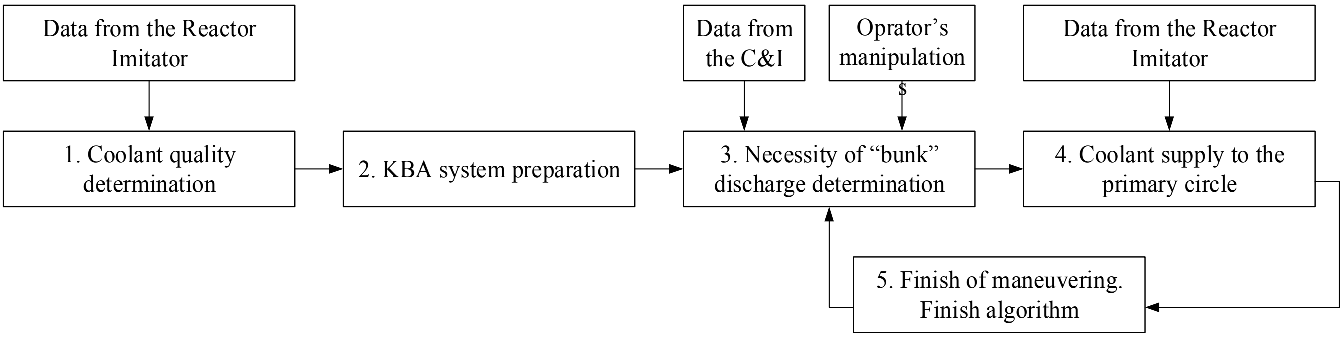

Figure 2. The flowchart of the KBA-KBC-1 system control algorithm

FGC gives commands to equipment and/or subordinate algorithms that are lower in hierarchy, for example process protections and interlocks, automatic transfer switch automatic control system, and it monitors the conditions for next step switch-over. The flowchart of developed KBA-KBC-1 systems control algorithm is shown at the fig. 2

During the maneuver dynamic processes the reactor's neutron flux changes greatly, so it leads to changing of the other neutron and thermohydraulic characteristics of the core and another equipment. It means that such control algorithms design calls for the precise forecast of these physics characteristics. As the source of these forecasts, the "Reactor Imitator" (as in the text – RI) code was used for KBA-KBC-1 systems control development. The RI code designed for VVER-1200 reactor mathematical modeling and it used at the operating nuclear power plants as a reliable information support for reactor operator. Information support is available in regime "on-line" of RI working. This regime presents the following information:

– the latest reactor state – the neutron and thermohydraulic characteristics before calculation starts;

– the current reactor state – the neutron and thermohydraulic characteristics which are measured by Internal Reactor Monitoring System (as in the text – IRCS);

– the critical boric acid concentration and axial power tilt prediction under the influence of xenon effect.

Mathematical functions that describe the changing of these values over time are used for prediction. Exponential function is selected for critical boric concentration prediction, and sinusoidal function for axial power tilt. The function's factors are determined by experience-based data.

Mathematical modeling in RI is available with "off-line" mode and it runs independently because it does not rely on current reactor state. Operator inputs initial data, either way they could be read from data files. Input data files can be created by monitoring the real reactor state when RI is working in "on-line" mode.

III. Algorithm Description

Description of the designed FGC algorithm is below.

Available Modes:

– Manual;
– Auto.

Execution Direction:

– Start;
– Finish.

Starting Initial Signals

– from operator (priority 3);
– automatic (priority 2);
– protections and interlocks (priority 1).

Starting Condition: more priority signals of opposite directions are off and permission signal for required direction exists.

Repeat Starting Condition: if algorithm achieves the latest step of chosen direction (start-up/shut-down), repeat starting is available only after precursory switch to "manual" mode.

FGC GROUPS OF STEPS

Group 1. Determination of required coolant type for delivery to the core.

According to the Energy Operator or planned power system load curves reactor operator is bound to switch the FGC algorithm to start-up direction in auto mode if it is possible to obtain data from the RI. Depending on the initial data received from RI, algorithm runs to delivery boric acid or pure condensate. When signals from RI are collected, the command to control of KBC system valves (in case of pure condensate delivery) or JNK (boron solution storage) system valves (in case of boric acid is required) is sent. Then the KBC system preparation for required coolant supply to feed pump suction is going on.

Group 2. Preparation of KBA system

The purpose of this group of steps is transport time reduction of boric acid and pure condensate delivery to reactor core down to 7-10 minutes. It is now possible by engaging additional KBA pump with low flow rate, so that expected total flow from two pumps exceeds 18 m^3/hr.

Group 3. Necessity of "bunk" discharge determination

For required starting coolant supply, it is necessary to make sure that coolant in the pipeline section from the KBC system pumps discharge to KBA pumps suction (this pipeline section called "bunk") has the right boron concentration. If this concentration varies from target value, it is necessary to remove this coolant volume by draining it to KTC (boron drainage) system tanks. There are two ways to "bunk" draining:

1. *Auto* – control is providing according to boron concentration meter indicated values. Conditions are presented with the (1):

$$\begin{aligned} C^{required}_{H_3BO_3} &= C^{bunk}_{H_3BO_3} - not\ draining, \\ C^{required}_{H_3BO_3} &\neq C^{bunk}_{H_3BO_3} - draining \end{aligned} \qquad (1)$$

2. *Manual* – operator controls the process.

Result – replacement of the coolant in the "bunk" with the required concentration coolant (pure condensate or boric acid).

Group 4. Coolant supply to the primary circle

Required coolant supply to the reactor until follow condition is achieved (2):

$$F_{H_3BO_3/\,pure\ condencate} = 0\ m^3\ /\ hr \qquad (2).$$

When this condition is achieved, FGC waits for signal from RI for algorithm continuation or finishing.

In case of algorithm working continuation, FGC returns to group 3.

Group 5. End of maneuvering. Finish algorithm

When the FGC gets the finishing signal from RI, it forms the command to switch the KBA-KBC-1 systems equipment to "ordinary operating" mode.

IV. CONCLUSION

The FGC application is function set which receive the initial data from C&I, RI and/or operator. It interprets the data into control signals and, in this way, the algorithm provides the equipment control and changing between maneuver and ordinary modes. Using the FGC allows to increase automation level during the neutron power control and, due to step-by-step algorithm following with their indication at the DCS (Digital Control System) monitors, to generally develop the nuclear safety of dynamic modes which is required during the nuclear-hazardous tests conduction.

The proposed engineering project implementation will allow nuclear power plant unit to operate with wide daily power output range and keep safety and efficiency quality at high standard.

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Digitizing the last stage of the design documentation development

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I. INTRODUCTION

Rosatom production system (PSR) — a systematically coherent complex of the processes. PSR comprises operational principles, rules, management tools which give rise to continual improvement. PSR is logically completed vision of an effective process management. The main idea of PSR is reducing different kind of negative profit. Senior executives have an opportunity to create a project if there is an improvement validation of it. When a project was started some employees become an integrated process team. Each member of team specializes in a different field of work. The main idea of this action is that an issue can be solved more efficiently by employees who face with the issue daily. Furthermore project team members try to minimize non-value-added effort and substitute it with profitable work. Owing to these changes, a working process turns in an efficient way.

One of many PSR projects was about optimizing the design documentation compliance assessment process. The final stage of the design documentation development, in particular, the compliance assessment, is a multistage process [1]. Design documentation has been being developed with using computer-aided designing at JSC «Afrikantov OKBM» for more than 15 years. Owning to these facts the compliance assessment process should be digitized. The project was opened in 2017. The boundaries of the project included more than 20 design departments, the Design Department of Standardization, the Department for Developing Product Lifecycle Support Systems and the Documentation Circulation and Reprography Department.

II. HOW THE PROJECT WAS REALISED

The main goal of the project was to reduce the time taken for the design documentation compliance assessment process. For achieving the goal, a number of tasks had to be solved. Also a sequence of actions, which leading to a result, had to be drawn up.

There are major tasks which were solved as part of the project:

– identification the issues;
– identification a root cause of long time taken for the process;
– finding optimal solutions and testing them [2].

Firstly, studies were conducted to obtain information from the stakeholders with using one of PSR tools – «Questionnaire No. 1». «Questionnaire No. 1» is a list with questions about current working process. These studies allowed collecting independent opinions about the process running and suggestions for the process improvement.

Secondly, an integrated process team used a PSR tool called «Process current status map». It needs to determine all stages of the process and identify the stages with issues.

After «Process current status map» was prepared, there were seven stages of the compliance assessment process. There were stages which included a hardcopy design document circulation:

– an initial check on a document hardcopy version;
– responding to comments on a document hardcopy version;
– a repeated check on a document hardcopy version.

After all comments for a document hardcopy version were fully responded to, there were next three stages which included an electronic design document circulation:

– an initial check on a document electronic version;
– responding to comments on a document electronic version;
– a repeated check on a document electronic version.

After all comments for a document electronic version were fully responded to, the seventh stage of the compliance assessment was signing electronic and hardcopy versions of a document.

Then the stages which included the issues were defined:

– a repeated check on a document;
– responding to comments on a document.

The problems were identified too:

- the long time taken for repeated check on a design documentation to respond to comments;
- failing to respond to comments in the full scope.

The third point is using a PSR tool called «Production Analysis No. 1» to determine the stages which duration might have been reduced. Also, a root cause of the long time taken for a repeated check on design documentation to respond to comments was identified at the stage of collecting actual data. It was an absence of a regulated method of recording comments in the course of the compliance assessment.

Due to the fact that a corporate-level multipurpose data entity management system called IPS (Intermech Professional Solutions) is being used at the company, project team members decided:

- three hardcopy document circulation stages ought to be removed from the compliance assessment process through switching to an electronic document circulation;
- an electronic document circulation process with a function of distributing business processes from a compliance assessment team leader to employees ought to be organized;
- an initial check on documents ought to be made on the IPS program with using the «Red Pencil» function.

Before the electronic design document circulation of compliance assessment process was put into an industrial operation, there were a test version of the process and a pilot-industrial operation of the process. After three months of a pilot-industrial operation were over, «Questionnaire No. 2» and «Production Analysis No. 2» were conducted. Since the results of the studies were positive, project team leader arrived at a conclusion about rationality of putting the process into an industrial operation. Despite the fact that the project was closed in 2018, employees are able to create «Suggestion for improvement» for continual improvement of the process.

III. Conclusion

The accomplished work resulted in that the process time was reduced from 33 to 16 work days at average, which is evidenced by a comparative description of the results production analyses (Fig. 1). As a result there is a positive effect upon a fulfillment of contractual obligations on established dates. In addition it should be said that advent of technology allowed not only reduce the process time, but also create an informational heritage for next generation of engineers.

Performing the compliance assessment in an electronic form makes possibility to retain the comments on the design documentation along with the document in the IPS electronic space. This fact has a positive effect upon the issue called «failing to respond to comments in the full scope». Owning to the fact that the comments are represented with using a regulated method and always accessible for viewing. As remembering is a reconstructive process, it may produce errors. We may reconstruct events in the form that we prefer to keep the events in mind rather than in the form that we actually experienced them [3]. The visual details of an object, even a very familiar object, are typically available from memory only to the extent that they are useful in everyday life. It was also suggested that recognition tasks may make much smaller demands on memory than is commonly assumed [4].

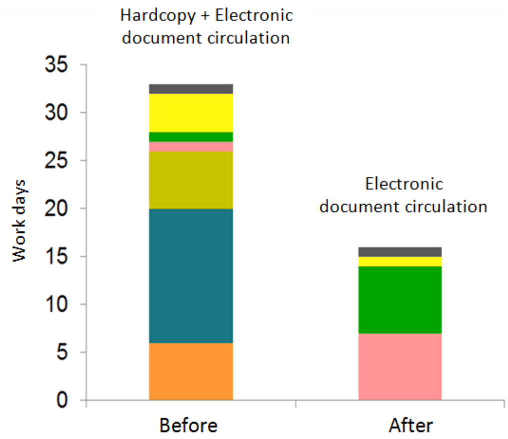

Figure 1. A comparative description of the results production analyses.

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Robotics and Automation In The Nuclear Industry

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I. Introduction

The use of robotics and automation in the nuclear industry has the potential to bring large improvements in safety, efficiency, product quality and reliability. This is particularly true in the fields of nuclear medicine and decommissioning, which often require the manual handling of dangerous material. However, there are also many barriers to the implementation of automated systems. Some of these are cost, maintainability, regulatory requirements and the need for the system to function in high radiation environments.

Opportunities for automation at ANSTO led to the completion of broad research into this field. Examples of robotic decommissioning in the US, UK and Japan where studied. A collation and comparison of "off the shelf" products available on the market was completed and various examples of automation that already existing at ANSTO where studied.

II. Nuclear Automation Challenges

A. Radiation Damage

Radiation causes damage to certain materials such as polymers, causing them to break down or become brittle over time. Radiation can also interfere with digital electronics, causing bit-flits or more permanent damage. Several strategies can be employed to combat these effects. Some of these include:

- Choosing radiation resistant materials
- Shielding vulnerable components
- Removing vulnerable components from radiation field (e.g. designing a hot cell robotic arm with all the electronics on the outside of the cell)
- Building systems with redundancy
- Replacing components on a regular basis

B. Maintainability

Another major concern is the maintainability of automation and robotic equipment. Given that maintenance tasks likely expose staff to radiological hazards, systems should be designed to be as easy to maintain as possible. Some examples of how to achieve this include:

- Making equipment easily accessible
- Designing the system to be modular so that broken components can be removed and replaced quickly
- Including manual overrides for electrically driven axes
- Designing components to be easily cleaned and decontaminated
- Building systems with redundancy
- Stocking spares to reduce downtime.

C. Cost

The final prohibiting factor is cost. Automation system and robotics can be very expensive. In the manufacturing industry, the use of robotics is economically justified because they are completing highly repetitive tasks, at a much faster rate than humans a capable of. These situations are not as common in the nuclear industry. Cost-benefit analyses often find that automation projects are not economically justified. To properly reflect the benefits of automation these analyses must consider if the project also results in an improvement to safety. This can come in the form of a reduced dose to staff, as well as a reduction in the likelihood of an accident occurring.

III. International Examples – Nuclear Decommisining

A. West Valley Demonstration Project

In the US, the decommissioning of the vitrification facility at the West Valley Demonstration Project was aided by the use of a remote-controlled mobile demolition robot. This robot could interchange the tool connected to its end-effector. Shears were used to snap jumpers, saws were used to cut through electrodes attached to a melter (see Figure 1), and a combination cutter-gripper-spreader was used to pick up debris [1].

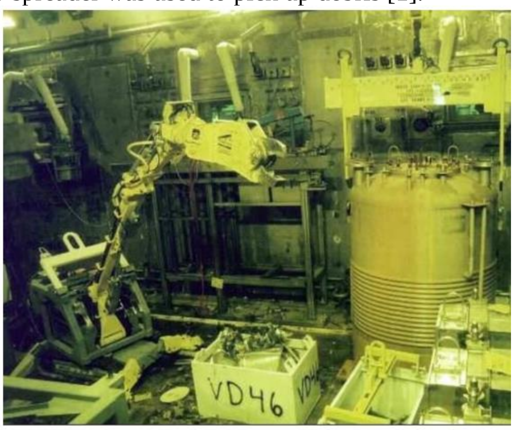

Figure 1. BROKK robot being used during decommissioning of the vitrification facility at the West Valley Demonstration Project [1]

B. Fukushima reactor

In 2013, Japanese company Toshiba unveiled a remote-controlled decontamination robot designed to be used at the Fukushima reactor (see Figure 2). The robot is designed to decontaminate surfaces by simultaneously blasting them with dry ice and vacuuming up contaminants [2]. While the robot is mobile and remotely controlled, it is large and heavy and difficult to maneuver into tight spaces.

Figure 2, Toshiba decontamination robot designed for the Fukushima Reactor [2]

C. Sellafield's Lasersnake

Sellafield's 2016-2017 End of Year Report on Decommissioning Capability Developments [3] featured a collaborative project called the Lasersnake. The Lasersnake is a high-powered laser cutter mounted on the end of a multi-jointed, snake-like, robotic arm (see Figure 3). The Lasersnake can enter a hot cell through a single port and once inside is highly manoeuvrable. Most of the electronics are located at the back of the robot arm, making the device resistant to radiation damage. In 2016 the Lasersnake was used in size reduction activities related to a redundant dissolver [3].

Figure 3. Sellafield's Lasersnake [3]

IV. Robotics at ANSTO

A. SyMo Waste Management Facility

The SyMo Waste Management Facility currently under construction at ANSTO employs a high degree of automation. The process of mixing liquid waste with the additive product and converting the resulting mixture into a dry powder is a fully automated batching process. The evacuation, filling and welding of the SyMo CANs, as well as the loading of these CANs into the Hot Isostatic Press (HIP) is also automated. This system will include a conveyor that moves CANs between stations and a computer vision system for identifying and storing information related to each CAN into a database. The SyMo design also includes a maintenance robot which will be located next to the HIP (see Figure 4). This robotic is a 6-axis KUKA robot with an interchangeable end effector and a rated payload of 360kg. The robot arm will lubricate a circular seal every time a new CAN is loaded into the HIP. The robot will also be able to dismantle certain components of the HIP in the event of a breakdown, reducing the amount of manual labour required from maintenance staff and therefore also reducing their radiological risk.

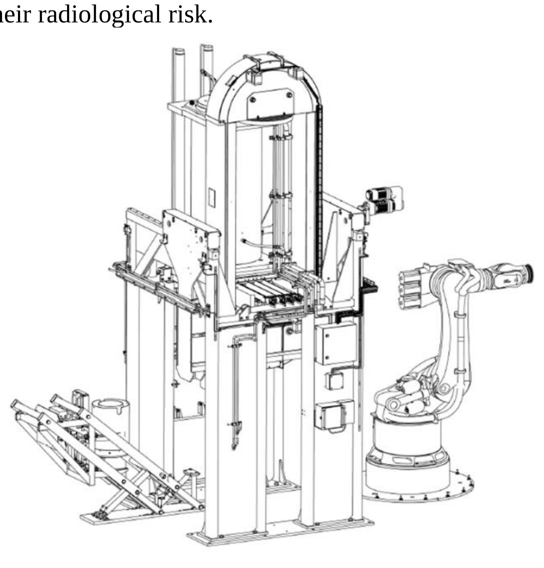

Figure 4, The SyMo HIP with the automatic loading mechanism on the left and the KUKA maintenance robot on the right.

B. Bulk Mo-99 Quality Control

At the time of writing, work is being completed to automate the sample preparation process for bulk Mo-99 quality control checks. The automation of this process has been broken up into three stages. Stage one, which is already complete, involved the design, implementation, and testing of a de-capping robot (see Figure 5). This machine automatically removes the cap off vials so that further steps in the QC process can be completed. Stage two involves making modifications to the existing fume hood in order to prepare for stage three. Stage three involves implementing a fractioning system to automate the sample preparation process. Comecer has provided ANSTO with a concept design of such an automated system. This design (see Figure 5) includes the automation of de-capping, micropipetting, diluting and the lowering of vials into the ionising chamber to check for activity levels. A rotating turntable is used to move vials and other components to the correct position during different stages of the process.

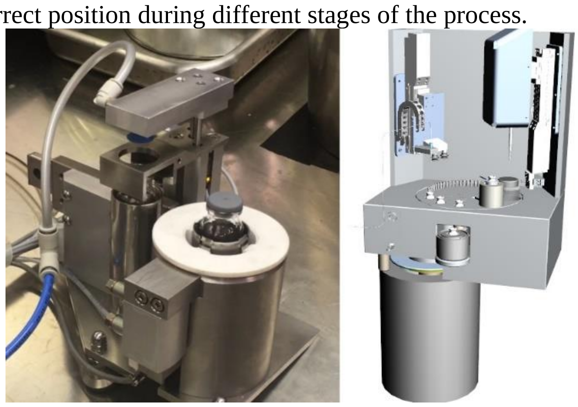

Figure 5, De-capping robot designed and built at ANSTO (left), 3D model of fractioning system designed by Comecer for ANSTO (right).

C. *Supercompactor*

The Supercompactor located ANSTO's campus in Lucas Heights, employs a high degree of automation. Barrels are first labelled and loaded onto a conveyor system manually. They are then automatically weighed and a gamma spec of each barrel is taken. Without ever needing an operator to be present, the system will automatically perform these actions for all the barrels loaded onto the conveyor. The system reads the labels on the barrels at each stage and the information is added to a database. The barrels are next loaded into the Supercompactor, punctured and the compaction process begins. These steps are performed automatically, although an operator is required to be present in the control room. After being compacted, the resulting pucks are automatically picked up by a gripper and mounted on a small gantry crane. The pucks are placed in a storage grid on the ground (see Figure 6). After a number of pucks have been stored, the operator can initiate the next stage in the process. The crane measures the height of all the pucks stored in the grid. The system runs an optimisation algorithm to determine how the pucks should be stacked in the overpackers so as to best use the available space. The crane then picks up the pucks and stacks them into the overpackers as determined by this algorithm.

Figure 6, Pucks stored in grid pattern by the xyz transfer system post compacting.

Acknowledgment

This research was supported by ANSTO. I thank Ben Biggrig, Brendan Mills and Lester Bemand for assistance with their technical input and expertise.

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High Flux Australian Reactor, A brief revisit of Australian's First Nuclear Reactor

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I. Introduction

The High Flux Australian Reactor (HIFAR) was one of only 70 reactors worldwide capable of producing in demand medical radioisotopes. Of British design, HIFAR is one of six DIDO-class reactors and Australia's first national research reactor. HIFAR was central to the research that occurred at ANSTO and operated safely and reliably for almost 50 years (1958 to 2007).

ANSTO enjoys a favorable reputation internationally, and HIFAR was at the center of much of ANSTO's early research work.

In the early years, the purpose of HIFAR was to test materials for nuclear power reactors. Over time HIFAR evolved to produce neutrons for the production of nuclear medicine and for scientific research. Being able to meet the majority of Australia's nuclear medicine needs it also found a purpose providing commercial services such as neutron transmutation doping of silicon, and sample analysis services for CSIRO, Australian Universities and other industries.

Neutrons are subatomic particles found in the nucleus of all atoms. HIFAR produced neutrons through the process of fission, the splitting of a large atom uranium, into two smaller ones; and one or more neutrons. Fission occurs when a heavy nucleus absorbs a neutron and splits. Some of the neutrons given off in the process of fission, after slowing down (losing energy), are used to keep the fission chain reaction going.

Some uses and benefits from HIFAR included:

a) Production of radioisotopes for medical purposes and for industry.

b) Silicon transmutation doping for the semiconductor industry.

c) Neutron Activation Analysis and Delayed Neutron Analysis for the mining industry and forensic purposes.

d) Neutron diffraction experiments for the study of matter.

ANSTO. *(Sponsors)*

Figure 1. HIFAR, Lucas Heights, 1958. *Australian News & Information Bureau.*

II. Uses and Benefits from HIFAR

A. Production of radioisotopes for medical purposes and for industry

Some radioisotopes (radioactive isotopes of an element) are highly valuable in the diagnosis of functionality of specific internal organs and the treatment of disease. When it comes to diagnosis, radioisotopes that typically have short half-lives are utilized; special cameras are used to capture the emitted radiation from radioisotopes once they have entered the human body. Using various imaging techniques the radioisotope distribution can be tracked as it circulates through the body, and will typically decay before causing damage to the body [1]. For treatment or therapeutic purposes the medical industry states radioisotopes with longer half-lives can be more suitable [2].

Medical radioisotopes are made from materials bombarded by neutrons in a reactor.

Within Industry, radioisotopes are commonly used to conduct stress testing or integrity testing of welds, measuring thickness of materials, or gauging liquid levels within containers. All of these methods utilise a gamma source [3].

B. Silicon transmutation doping for the semiconductor industry

Silicon irradiation also known as Neutron Transmutation Doping (NDS) is a process in which thermal neutrons react with the silicon atoms and by changing some of those atoms to phosphorus, the resistivity of the silicon is reduced. This thereby improves the silicon's ability to conduct electricity.

C. Neutron Activation Analysis and Delayed Neutron Analysis for the mining industry and forensic purposes

HIFAR's capabilities included the ability to determine the concentrations of elements in a material, which were applied in a multitude of research fields, such as medicine, environmental monitoring and forensic science. Neutron activation analysis is a method of analysis in which materials are bombarded with neutrons, which when absorbed by the matter, activate elements into a state leading to new materials. By monitoring the radioactive emissions from the sample material and using known radioactive decay chains, the ability to trace elements and precisely identify the concentrations of the elements within the material is achievable. Analysis can leave samples irradiated; being radioactive, they need to be handled and stored by properly trained personnel. [4]

Delayed neutron analysis follows the same principle as neutron activation analysis but sees the monitoring and measurement of the rate of emission from the material after it has been irradiated.

D. Neutron diffraction experiments for the study of matter

Neutron diffraction is the process of exposing a sample to a beam of thermal or cold neutrons and the intensity or diffraction pattern around the sample provides information on the structure of the material. Within HIFAR neutron beams were directed through narrow holes in the shielding called collimators and onto specimens. This would scatter the neutrons and the scattering is recorded by neutron counters. The technique would allow researchers to obtain information about the atomic structure of the materials [5].

III. THE DETAILS

The uranium core of the reactor was located in the center of a 2 metre diameter aluminium tank filled with 6 tonnes of heavy water (D_2O). A steel tank packed with graphite enclosed the aluminium tank, reflecting neutrons back into the reactor and conserving them. The steel tank was then encased in a 10cm thick lead shield and 1.5 metre thick dense concrete shield. These layers served to absorb neutrons and protect the operators from radiation exposure. A sealed, steel circular building 21m high and 21m in diameter provided containment and the iconic, unique look of HIFAR.

Twenty-five cylindrical fuel assemblies of approximately 10cm in diameter were spaced 15cm apart and vertically suspended into the aluminium tank of heavy water. Each fuel element consisted of concentric metal tubes 2mm thick that contained the uranium fuel. Each fuel element contained 283 grams of uranium. At initial operation fuel elements were enriched to >90%, but by 2007 it was down to <20%. Heavy water was pumped at high speed through the fuel assemblies, between the fuel element tubes to cool them. The heavy water's function was to slow down the neutrons which were liberated by fission in the reactor core. Six cadmium arms approximately 1.45m in length were positioned and moved in an arc between the rows of fuel elements as necessary to determine the reactor power and shut it down as needed.

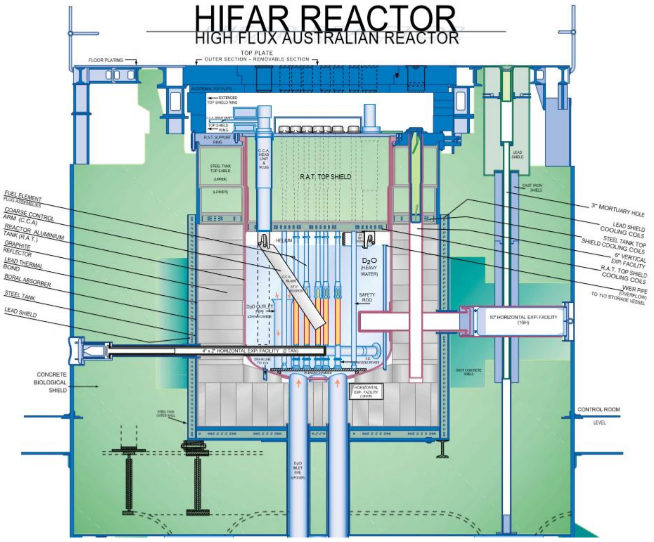

Figure 2. General Arrangement of HIFAR Reactor, Sectional Plan. *ANSTO.*

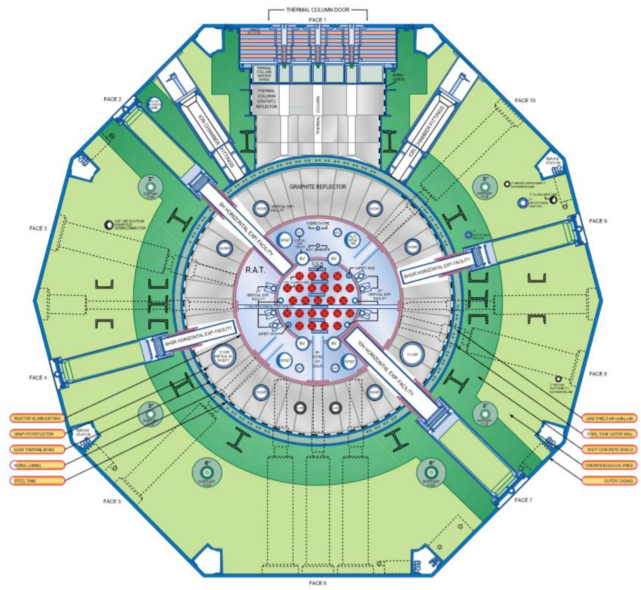

Figure 3. General Arrangement of HIFAR Reactor, Sectional Plan. *ANSTO.*

IV. HIFAR LIFECYCLE STAGE

A. *Care and Maintenance Phase*

HIFAR is currently in a period of Care and Maintenance, with Facility Personnel ensuring it is maintained according to the requirements set out in the Possess and Control License issued by the CEO of the Australian Radiation Protection and Nuclear Safety Agency (ARPANSA).

A crew much smaller than the Facility once employed when operating, now retain a vast amount of the knowledge gained from operating and maintaining HIFAR over its lifetime. Current Care and Maintenance tasks ensure the required safety and auxiliary systems are inspected and maintained regularly. Of the six DIDO-class reactors built globally, none have yet to be fully decommissioned. HIFAR was the last to cease operation in 2007.

B. *Characterisation*

ANSTO has supported the Characterisation efforts of HIFAR since its shutdown in 2007. An ANSTO team will release a report outlining the findings of their characterisation efforts of HIFAR in the near future that may provide key insights into the decommissioning phase.

C. *Decommissioning*

HIFAR will eventually transition from Possess and Control to a Final Site Clearance which will enable decommissioning to take place. This will involve preparing decommissioning plans, preparing and submitting re-licensing documentation of the Facility for decommissioning, then the physical dismantling of irradiation rigs, experimental equipment, and instrumentation, removal of remaining wastes and/or hazards for transportation to a waste holding facility, and an array of other safety and administrative activities. The quantity and type of waste decommissioning will generate and current processing and storage capabilities have to be taken into account when determining the timeline and decommissioning plans for HIFAR. While the Australian Government has been implementing the process of creating a National Radioactive Waste Management Facility, physical decommissioning may have to take a back seat until suitable waste holding facilities are made available.

ACKNOWLEDGMENT

"J.G thanks ANSTO for the opportunity to learn about the High Flux Australian Reactor and shadow personnel that were present when HIFAR was operational as well as current caretakers".

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SYDNEY

Thermal Neutron Guide In-Pile and Shutter Replacement: Key Design Improvements

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I. Introduction

After several years of planning and preparation, ANSTO (Lucas Heights, NSW, Australia) will have an extended shutdown in 2020 to replace the thermal neutron guide in-pile, shutter and front cover (Fig. 1) adjacent to the OPAL (Open Pool Australian Lightwater) reactor.

The in-pile houses guides with reflective supermirrors on the internal surface that allow neutrons to travel to multiple neutron beam instruments. The shutter also contains guides surrounded by shielding and is able to rotate to block neutrons from travelling downstream. The front cover seals the in-pile and shutter inside the wall cavity and acts as the secondary loss of coolant accident barrier for the reactor. The <1mm thick aluminium foil windows on the front cover have been hydrostatically tested to remain sealed in the event of a coolant leak but still allow neutrons to travel through under normal operating conditions.

The periodic replacement of these components is necessary as the deterioration of the reflective supermirrors from high levels of radiation negatively affects the neutron beams and scattering activities used by ACNS (Australian Centre for Neutron Scattering).

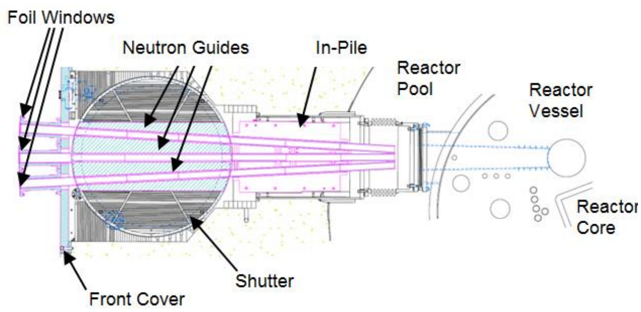

Figure 1. Neutron Guide Assembly – Schematic Plant View

The original design of the OPAL reactor allowed for three thermal neutron guides but ultimately only two were installed. This replacement will increase the thermal neutron guides to three. The additional guide will be extended at a later date to be used by ACNS for additional neutron beam instruments.

In 2012, ANSTO successfully completed its first in-pile and shutter replacement for the cold neutron guides in a five-week long extended shutdown. The lessons learnt have been incorporated to refine the design and planning for this project. The following paper will provide an overview of some of the main design improvements.

II. Design Improvements

A. Neutron Guides

A key design improvement was to the neutron guide substrate material. The original neutron guides were manufactured from optifloat glass substrate with a supermirror coating, which has a design life of 10 full power years (~300 days per year). During the cold neutron guide replacement, damage to the neutron guide supermirror coating was observed in guide glass more than 4 metres downstream (Fig. 2). It is expected that guides in the thermal in-pile and shutter, which have experienced higher neutron flux, would have much more extensive damage.

Figure 2. Neutron guide damaged observed in 4 metre+ downstream guides

Due to improvements in neutron guide manufacturing technology, the new replacement guides have been manufactured from aluminium substrate (Fig. 3). Aluminium substrate guides are being used worldwide and are anticipated to last much longer than glass; potentially, they could last the remaining life of the OPAL reactor. Literature reviews indicate that the radiation damage for aluminium structural components would commence only with neutron flux levels of around 10^{22} n/cm^2 for fast neutrons and at around 10^{23} n/cm^2 for thermal and epithermal neutrons [1]. Given that the OPAL reactor experiences much less neutron flux (10^{12} n/cm^2/s) in the first 6cm of the guides, this suggests a much longer life when using aluminium substrate.

Figure 3. Aluminium substrate neutron guides

Another aspect of damage to the neutron guides is the peeling of the supermirror from the substrate due to radiation. Tests performed by SwissNeutronics, the supplier of the neutron guides, indicated the tendency for such peeling was much more related to the glass substrate compared to aluminium.

ANSTO conducted a risk assessment comparing aluminium and glass substrates which considered the potential of damage (mentioned above), neutron activation, heat load and seismic effects. It was found that the option to use aluminium substrate resulted in medium-high risks, which were mainly associated with the damage to the guides causing a significant reduction in neutron flux. However, these same risks still exist for the glass substrate. The advantage of the glass substrate was simply wider operational experience, hence greater confidence in its expected design life. Ultimately, ANSTO has opted to use aluminium substrate based on test records and operational experience at other establishments such as NIST, which has shown success with using aluminium substrate for neutron guides over a 4-5 year period.

B. Shielded In-Pile Carrier

During the cold neutron replacement project in 2012, it was discovered that the linear bearing assemblies had collapsed and their components were scattered throughout the inside of the wall cavity. A minor misalignment between the in-pile cavity and the shielded in-pile carrier rails may be a reason for the linear bearing collapse. The linear bearings are a high precision component and a misalignment of 1mm could potentially result in their collapse.

As a result, the in-pile could not be extracted into the shielded in-pile carrier with the intended pull tool. A come-a-long winch set-up (Fig. 4) was used to remove the in-pile, storing it temporarily inside the shielded in-pile carrier. For final storage, the rails on the shielded in-pile carrier needed to be heavily oiled for the in-pile to be removed.

Figure 4. Come-a-long winch set up for cold neutron guide in-pile removal

It is expected that the thermal neutron replacement project will face the same challenges. The project has developed an additional gearbox on the shielded in-pile carrier (Fig. 5) to improve the method of both removal and installation of the in-pile into its final storage location. The in-pile will be removed with a high-precision leadscrew secured to the gearbox via a ball screw support bearing and to the in-pile through an existing tapped hole (Fig. 6). During insertion of the activated in-pile into final storage, a similar method will be used; however the storage area has restricted horizontal space. Highly accurate modifications were made to the leadscrew to separate it into various segments which still maintained a seamless pathway for the ball screw support bearing rollers to travel on when re-connected. These improvements are anticipated to reduce strain, time and dose to technicians.

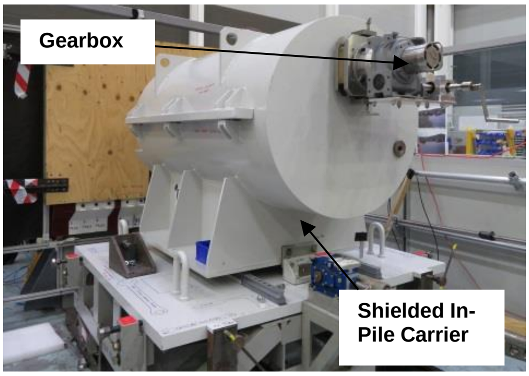

Figure 5. New gearbox mounted to rear of shielded in-pile carrier

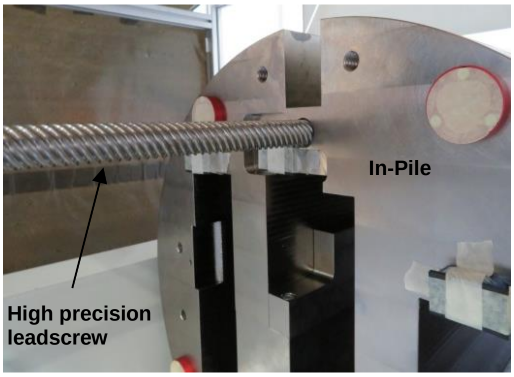

Figure 6. High precision lead screw for in-pile removal

As mentioned, a possible cause of the linear bearings collapse may be due to misalignment between the in-pile cavity and shielded in-pile carrier rails. To address this, another radiation-proof camera will be mounted on the shielded in-pile carrier which will capture the engagement between the two rails and check for any visible misalignment. The engagement between these rails has been repeatedly practiced on a mock-up (Fig. 7) which is identical to the thermal neutron guide wall cavity. This practice allowed for determining an ideal camera angle but also familiarised technicians with the tactile feedback of an ideal in-pile extraction.

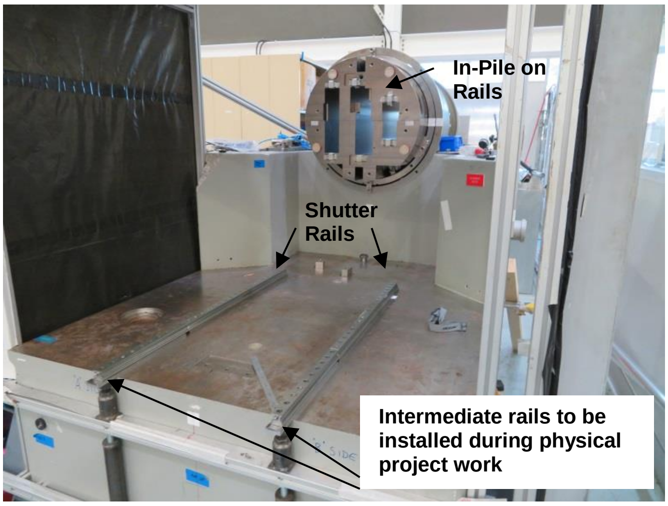

Figure 7. Mock-up of the wall cavity

C. *Removal of loose components inside the wall cavity*

The installation of the new in-pile requires the cavity to be thoroughly cleaned and examined. Following the ALARA (as low as reasonably achievable) principle, an emergency vacuum tool with a live feedback radiation proof camera (Fig. 8) was designed and manufactured within the time constraints of the 2012 cold neutron guide replacement project to remove the highly active bearing components from a distance and behind shielding. It is imperative that all loose components are removed to prevent damage and to ensure that the new in-pile and shutter can be safely installed.

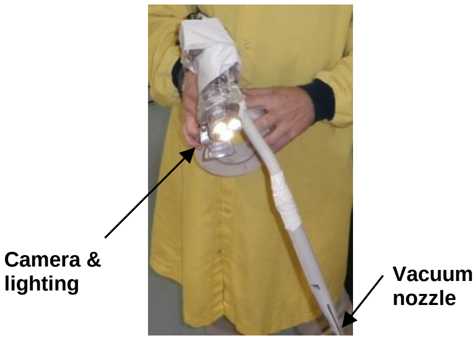

Figure 8. Emergency vacuum tool made during the cold neutron guide replacement

A streamlined vacuum tool equipped with a high resolution and radiation-proof camera with variable lighting (Fig. 9) has been developed and tested with technicians. The T-shaped aluminium tool allows technicians to manoeuvre the tool from behind shielding and at a distance to reduce leaning over into the beamline. The camera feedback will be mounted on the shielding to provide further improved ergonomics for technicians. Various 3D printed vacuum attachments have been tested to reduce the strain on technicians. A special lead pot has been designed to fit inside an off-the-shelf workshop vacuum cleaner to contain the loose components whilst still maintaining an adequate amount of suction.

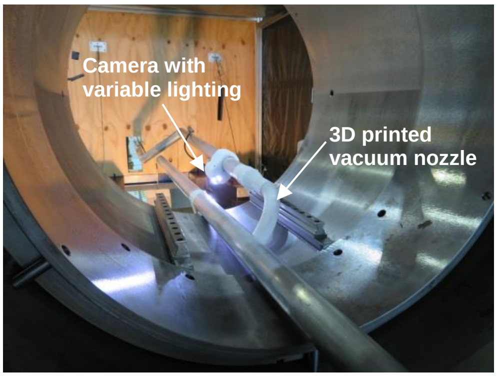

Figure 9. Improved vacuum tool during testing in the mock-up

III. Conclusion

The thermal neutron guide replacement will be the second guide replacement project undertaken since the commissioning of the OPAL reactor. ANSTO gained invaluable knowledge from the first replacement and have made key design improvements to the neutron guide substrate, shielded in-pile carrier and vacuum tool, which have all proven through literature or testing that they will be beneficial for the project. Other smaller improvements have been identified and implemented throughout training and mock-up practice.

In the coming months, more training and practice will be conducted on the mock-up to further prepare and refine the proposed installation procedure. Small improvements which keep in mind the ALARA principle will no doubt be made during these sessions. This project is sure to identify more lessons learnt to further improve the next neutron guide replacement currently scheduled for 2022.

Acknowledgment

Thank you to all staff across different areas of ANSTO for dedicating their time and utmost efforts on preparing for this major replacement project. A special thanks to our project manager, Kristian Veronika for overseeing all aspects of this major project. Another special thanks to Walter Bermudez for sharing his immense knowledge about the design and original commissioning of the in-pile and shutter; allowing us to better prepare and face challenges during the physical replacement process.

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TRACK 2: ADVANCED NUCLEAR SYSTEMS AND FUSION TECHNOLOGIES

ON-LINE REACTOR STUDIES AND RESEARCH OF NON-OXIDE NUCLEAR FUELS

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CORRECION FACTORS FOR REACTIVITY MEASUREMENT IN ADS

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DIGITAL ENGINEERING DESIGN PLATFORM DATA ANALYSIS AND MACHINE LEARNING ALGORITHM

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PRELIMINARY ASSESSMENT OF DECAY HEAT REMOVAL SYSTEMS IN THE ESFR CONCEPT: THE ROLE OF NATURAL AIR CONVECTION AROUND STEAM GENERATORS OUTER SHELLS IN ACCIDENTAL CONDITIONS

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THERMAL-HYDRAULICS OF LOSS OF HEAT SINK ACCIDENT OF INDIAN TEST BLANKET SYSTEM IN ITER

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DEGRADED CORE RELOCATION IN SODIUM-COOLED FAST REACTOR SEVERE ACCIDENT – PARTICLE-SIZE DEBRIS FLOW

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INNOVATIVE TECHNOLOGIES AND SAFETY ISSUE ON THE EXAMPLE OF THE WORLD'S FIRST FLOATING NUCLEAR POWER PLANT (FNPP)

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PROSPECTS OF CARBON-FREE ELECTRICITY DELIVERING CIVIL SUBMARINES FOR DISASTER RELIEF

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On-line reactor studies and research of non-oxide nuclear fuels

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I. Introduction

One of the important tasks of nuclear energy today is the formation of a new technological platform based on a closed nuclear fuel cycle with fast neutron reactor facilities that meet the principles of natural safety, make it possible to efficiently use nuclear fuel and reprocess the accumulated "tailings" of nuclear energy. This project is implemented in Russia by leading industry scientists and specialists and called the “Poryv” [1–2].

The development of safe and reliable reactor facilities necessitates the solution of a number of complex scientific and technical problems related to justification of safety and resource of fuel rods in normal, transient and emergency conditions.

To support operability of the fuel rod and ensure the safe and progressive evaluation of reactor performance, it is necessary to conduct reactor and non-reactor experiments with the study of the main resource-determining processes in a fuel rod: physical and mechanical properties, fission gas release (FGR), swelling and creep rapture, materials compatibility, evolution of fuel microstructure, stress - deformed shell condition, etc.

All this determines the relevance of off-site and on-site research to obtain experimental data on fuel and structural materials properties, and predict radiation behavior and deformation behavior of fuel rod.

The practical implementation of new reactor technologies imposes additional requirements on fuel rods being developed: increasing the service life; burnup; heat conduction; retention of fission products inside fuel rods; maintaining performance at high loads; manufacturability and profitability [3-6].

Therefore, dense fuel: mixed uranium and plutonium nitride (UPuN), dense nitride (UN) and carbonitride fuels (UZrCN) is considered as a promising fuel material capable of achieving a high burn-up depth (more than 15% at) with a high density of fissile actinides per unit volume and high thermal conductivity [4-10].

Despite considerable experience in the technological and experimental development of the materials [10-16], the amount of experimental data during irradiation, especially with respect to the fast neutron spectrum, is quite limited. Positive experimental research and available data on the internal reactor behavior of fuel elements with nitride fuel (FENIX, BR-10, BORA-BORA reactor experiments) cannot be used in full to justify service life of fast reactor fuel elements. In addition, there have been cases of increased gas swelling caused by increase in FGR as a result of nuclear reactions involving nitrogen and the formation of hydrogen and helium, as well as a violation of the fuel rod tightness due to the low creep rate and uncontrolled swelling of fuel [8-16].

Therefore, today there is an urgent need to study the behavior of non-oxygen fuel of a fast neutron reactor under irradiation. To obtain certified experimental data, it is necessary to develop methods and conduct reactor and non-reactor studies with the ability to measure properties in the research process under well-controlled conditions [10,12,16,17].

II. Methods

Due to the high cost and technical complexity of full-scale and loop reactor experiments, there is an urgent need for ampoule tests, which allow to recreate the necessary conditions for studying fuel behavior (including fission gas behavior during reactor test) and obtain reliable experimental data for verification of codes and models.

The analysis of existing developments and methodological approaches to the design of ampoule and loop channels for the study of fuel rods in research reactors is carried out. It is shown that experimental study of fuel radiation behavior is based mainly on postirradiation examination and therefore does not provide complete information about the kinetic parameters of radiation behavior.

To obtain a qualitative and quantitative description of the physical processes that occur with fuel elements directly in the process of irradiation, under limited time conditions, ampoule channel (AC) are proposed that allow to study the swelling and FGR on-line [18-25].

Such a model of ampoule reactor experiments with well-controlled conditions is the most effective and economically feasible way to obtain data on the fuel radiation behavior and deformation behavior of fuel elements. The developed AC are capable of ensuring the accuracy and stability of the irradiation parameters, and allow studying the kinetics of FGR and swelling during the irradiation process, as well as providing the possibility of easy defect detection and post-irradiation examination of the structural, physical, and mechanical properties.

Instrumentation AC consists of ionization chambers for gamma-ray detection and the primary helium purification system.

The sampler has inner volume, through which the helium gas passes. The type and activity of radioactive fission gas in the grab sample was identified by the energy analysis system using gamma spectroscopy. Sampling measurements were performed once every two days. The release rate-to-birth rate ratio, (R/B), is an important measure of the performance of the fuel. Fractional releases of short-lived fission gases can be expressed in terms of the (R/B), because the radioactive equilibrium is established quickly in the primary coolant circuit under normal operating conditions. The detected fission gas nuclides in the channel coolant of are Kr-85m, Kr-87, Kr-88, 133Xe, Xe-135, Xe-136, Xe-137, Xe-138.

The test process provides:

- continuous recording of fuel element operability parameters (temperature, energy release, neutron flux, FGR, etc.);
- comparative studies of FGR rate (R/B) from fuel materials, which is associated with fuel deformation during irradiation. The amount of FGR during irradiation (Kr, Xe, I) is analyzed by gamma spectrometry in the on - line mode;
- obtaining data on swelling and structural changes in fuel during irradiation;
- investigation of the complex stress-strain state of the fuel cladding.

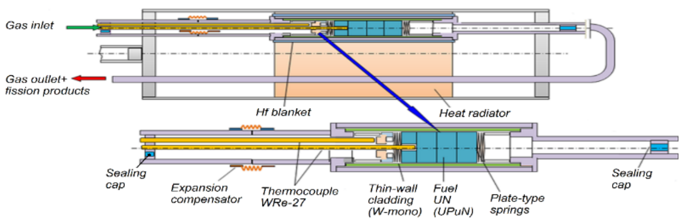

Figure 1. Ampoule channel for UN irradiation

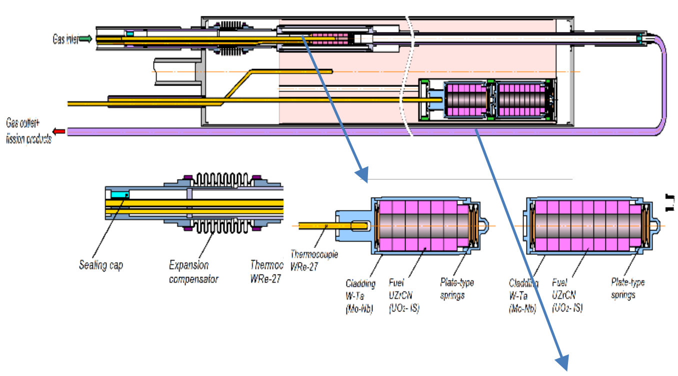

Figure 2. Ampoule channel for irradiation of different fuel and models of fuel elements

The developed AC (Fig. 1-2) are unified and suitable for testing various types of fuel. With the same circuit diagram of the AC, the design of the working areas with samples varies depending on the test parameters. AC also allows to conduct simultaneous testing of fuel rods with various design and technological features, under identical conditions.

The problems of limiting the dimensions of the channel, placing the required number of measuring sensors, preparing and comprehensively substantiating the operability and safety of AC are solved. AC are equipped with temperature and neutron sensors, pressure gauges at the inlet and outlet of the gas - main (to control the pressure of the carrier gas in the cavities) and allow to carry out reactor experiments under conditions close to the standard operating conditions (neutron flux, time, irradiation, temperature)

The simplicity of the AC design allows to use it without specific requirements for bench equipment as high-pressure lines. Another equally important feature of AC is the development of manufacturing, transportation and assembly technologies that ensure the integrity and tightness of samples up to loading into a research reactor, which is especially important for non-oxygen fuel and meets the basic principles of integrated safety.

At preparation stage calculation justifications will be carried out (thermophysical, neutron-physical, strength, radiation calculations), tests of the thermal package (determination of the heat and neutron-physical parameters of the research reactor), as well as non-reactor testing of materials and structural elements: mechanical studies, thermophysical and thermodynamic properties of materials. In addition, in non-reactor conditions, resource-determining fuel element processes are simulated: the influence of temperature on changes in the fuel structure and compatibility of materials [26-31].

The operability of AC was confirmed by reactor tests of fuel materials (UN, UZrCN, modified UO_2) and model fuel elements in the IVV - 2M research reactor:

- In 2012–2013 irradiation of UN fuel was carried out with determination of FGR kinetics and control of geometric dimensions at temperatures up to 1700 °C for a duration of 2500 h. The design of the AC provides parameters close to the standard operating conditions of the fuel elements [18-21].
- In 2016–2017 irradiation of fuel and cladding materials of the thermionic reactor (modified UO_2, UZrCN, W-Ta, Mo-Nb) was carried out with determination of FGR kinetics and control of geometric dimensions at temperature 1200 - 2000°C and duration of 8000 h. The AC allows simultaneously testing fuel compositions of various types in swept ampoules and model fuel elements under well-controlled conditions [22-25].

Reactor tests of promising fuels materials and model fuel elements showed the operability of the developed devices and the reliability of the selected design and technological solutions.

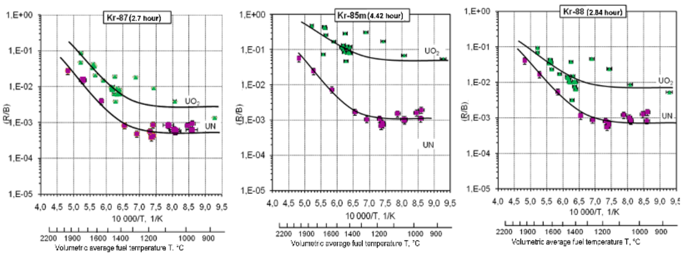

Figure 3. Relative yield of Kr from different fuel compositions

According to the results of reactor tests, it is confirmed that the implemented experimental concept allows:

- to obtain the dependences of the free swelling rate and kinetics of FGR depending on fuel structure and inventory, as well as on the burnup depth and irradiation temperature;
- to predict fuel behavior in on-line mode under different operation conditions of the nuclear power plant (in a wide temperature range (1200-2000) ºC and linear power up to 900 W/cm);
- to study the deformation behavior of fuel rod and the compatibility of fuel and cladding materials.

III. Results and Discussion

Kinetic parameters characterizing the gas swelling of the fuel were first obtained in the result of the experiments. It is noticeable that the rate of FGR (Kr-85m, Kr-87, Kr-88, Xe-136, Xe-137, Xe-138) from UN and UZrCN is much lower than from UO2 (Fig. 3). The obtained experimental data on the FGR kinetics confirm the processes of structure rearrangement that occur under irradiation.

An increased of FGR from modified UO_2 is associated with the formation of a columnar structure with open porosity and through radial microchannels along the grain boundaries. Gaseous fission products, which formed during the irradiation process, are unobstructed withdraw from the fuel matrix. This process can significantly reduce fuel swelling.

Low rate of FGR from UN is explained by a significant proportion of intragranular porosity, which retains gaseous and volatile fission products (with a small burnup of up to 3%, i.e.).

The results of comprehensive post-reactor studies confirm the reliability of structural and technological solutions developed by AC and allow us to talk about the safety of using the AC. Regular monitoring of geometric stability and tightness of fuel rods confirms the safety of the operation of the AC.

All the structural elements of the AC after irradiation retained their integrity and performance. Work parts of channel and capsules appeared without damage. Fuel elements retained their integrity.

IV. Conclusion

The developed AC devices make it possible to carry out comparative reactor studies and obtain continuous information during the irradiation process, especially necessary at the R&D stage, because they allow to form the most complete picture of the radiation behavior of fuel and the deformation behavior of fuel elements.

Thus, using the developed methods and devices, experimental data were first obtained on the kinetics of FGR and swelling of UN, UZrCN and modified UO_2. The results of studies of the kinetics of FGR during fuel irradiation show that the rate of FGR from UN is almost 4 times less than from modified UO_2 at equal temperatures. This is explained by the closed nature of the porosity of nitride fuel and the rather small achieved burnup values, which, apparently, are shorter than the incubation period of FGR. The obtained data allow us to describe the dynamics of gas swelling and conduct a comparative analysis of the behavior of fuel elements under irradiation.

The main advantages of using the developed AC are the efficiency and economic feasibility of the experiment (the ability to conduct tests in one cell in a short time).

The experimental data obtained during the irradiation process, necessary for the creation and verification of new-generation settlement codes, the improvement and optimization of fuel rod manufacturing processes, the safety justification and the reliability of the developed nuclear power plants.

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Correcion factors for reactivity measurement in ADS

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I. INTRODUCTION

The Accelerator Driven Systems (ADS) are one of the novel reactor designs proposed as part of partitioning and transmutation (P&T) of spent nuclear fuel. The three main benefits of such approach are limited overall volume of waste, limited required storage time and better fuel economy. Some P&T technologies are used at the industrial scale today, however they are mostly limited to reusing plutonium in form of the MOX fuel in regular light water reactors. Achieving fully closed fuel cycle in which all transuranic elements will be reused requires development of new reactor technologies.

Main challenge in transmutation of minor actinides is their negative impact on safety parameters such as reactivity coefficients and delayed neutron fraction. Since ADS is working in the subcritical state it leads to enhanced safety properties by increasing distance to prompt supercriticality. This enhanced safety margin can compensate for disadvantageous proprieties of minor actinides and leads to more efficient transmutation by allowing their higher concentration in the fuel [1].

For mentioned safety reasons core reactivity has to be known at any time during operation of the reactor, which is relatively new requirement, not seen in critical reactors used to date. The key requirements for utilized method are accuracy and robustness, but also simplicity since reactivity must be determined in real-time [2].

II. EXPERIMENTAL SETUP

The experiments described in this paper were performed at VENUS-F reactor at SCK-CEN facility in Mol, Belgium within European Project FP7 FREYA (Fast Reactors Experiments for Hybrid Applications). The VENUS-F reactor is the first zero-power mock-up of fast neutron spectra lead ADS. It uses metallic uranium fuel placed in the solid lead matrix. Use of lead makes its neutronics parameters close to those of possible future industrial ADS. The reactor core is a 12x12 square matrix, that can be filled with fuel assemblies or lead bars and surrounded by a lead reflector. Different core configuration are possible by using different loading patterns. It can be used both as critical and subcritical reactor. In the latter case D-T neutron source is placed in the center of the core.

The configuration used in discussed experiments was SC1 with $k_{eff} \approx 0.96$. Its radial cross section with detector positions (fission chambers) is shown in the Fig. 1. 93 fuel assemblies were used (marked violet in the figure) surrounded by lead assemblies (yellow) serving as the inner neutron reflector. The cylindrical lead structure surrounding whole core matrix called the outer reflector. There are also six safety rods (light blue) and two Control Rods (CR, red). The axial cross section is shown the Fig. 2.

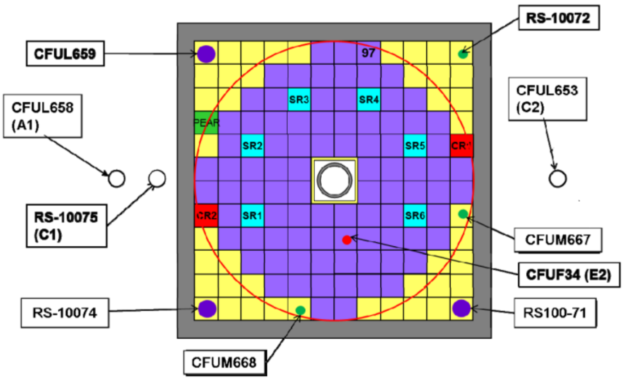

Figure 1. Radial cross section of the core in SC1 configuration with detector placement [4].

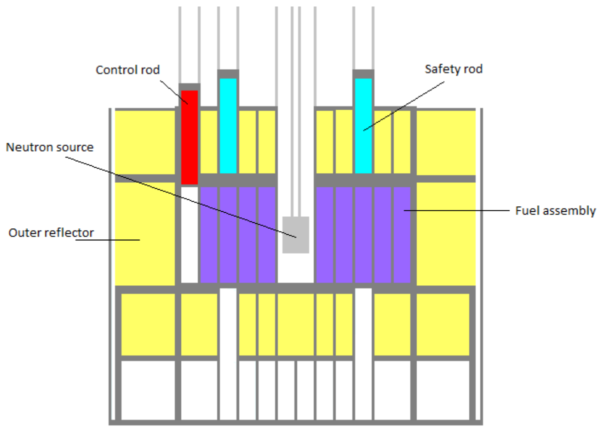

Figure 2. Axial cross section of the core.

When working in subcritical mode the reactor is driven by 14 MeV neutrons from D-T neutron source of the GENEPI-3C neutron generator . The GENEPI generator is able to work in three modes: pulsed, continuous and beam-trips (continuous with short, periodical interruptions of the beam) [3].

III. Preliminary Results

First set of experiments was done with the generator working in pulsed mode (600 ns long neutron pulses with 200 Hz frequency). The evolution in time of the neutron flux was measured after every neutron pulse from the source. It is shown in the Fig. 4 for selected detectors. One can observe that immediately after neutron pulse fast decay of neutron flux occurs which is then followed by period when it is constant. Reactor is then driven by delayed neutrons. Different rates of decay can be seen for different detectors. Is also should be noted that in case of the detector CFUL668 (and other placed in the outer reflector) full decay of prompt neutrons cannot be observed between the pulses. Basing on such neutron flux evolutions reactivity value can be determined using the Sjöstrand method [5]. In this method the reactivity is given by the formula:

$$\rho = -\frac{F_P}{F_d} \qquad (1)$$

Where F_p is the number of the neutron counts in the detector corresponding to prompt neutrons and F_d is the number of neutron counts corresponding to delayed neutron. Those fields are shown in the Fig. 3 which shows a theoretical neutron flux evolution in the reactor core after a neutron pulse from the source.

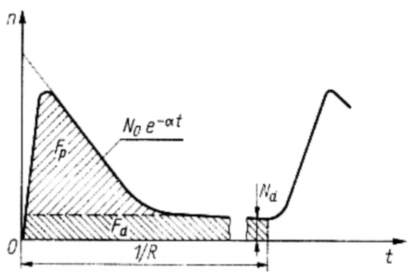

Figure 3. Neutron flux evolution after a pulse [6].

The reference value of ρ = -5.28(13) $ was used for core configuration SC1 with in position CR@479.3 mm. This control rod position corresponds to criticality point achieved in the critical core configuration and mentioned value was measured using MSM method [7]. For other CRs position reference value was calculated using MCNP code as distance from reference point mentioned before. All calculations were done using MCNPX 2.5 code with JEFF 3.1 data libraries.

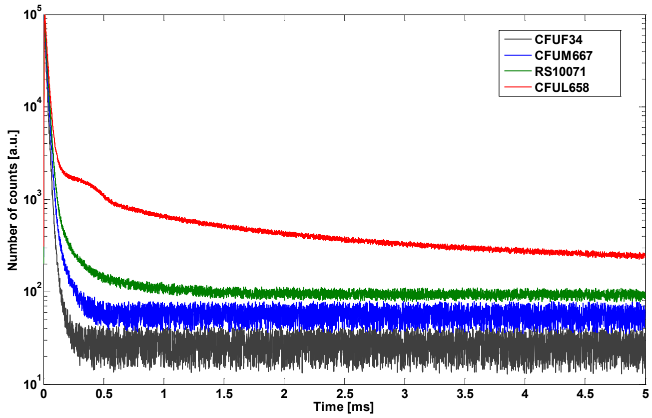

Figure 4. Neutron flux evolution after the pulse from the source - CR@479.3 mm

Obtained results are shown in the Table I. Graphic comparison of the results from different detector positions for selected case (CR@479.3 mm) is shown in the Fig. 4. One can see that obtained reactivity value strongly depends on detector placement in the system. Only a single detector (CFUF34) is showing reactivity values that agrees with the reference value, while others are showing values closer to criticality.

TABLE I. Reactivity values obtained with the Sjöstrand method

Detector	ρ [$]			
	CR@600mm	***CR@479.3mm***	***CR@240mm***	***CR@0mm***
CFUF34	-5.09(8)	-5.25(9)	-5.69(18)	-6.30(13)
CFUM667	-4.85(3)	-4.99(3)	-5.79(11)	-6.02(7)
CFUM668	-4.87(3)	-4.98(3)	-5.64(10)	-5.98(7)
RS-10071	-4.81(1)	-5.00(1)	-5.55(1)	-5.96(2)
RS-10072	-4.81(1)	-4.99(2)	-5.61(3)	-5.95(2)
RS-10074	-4.84(1)	-5.03(1)	-5.62(3)	-5.98(2)
CFUL659	-4.75(1)	-4.92(1)	-5.50(1)	-5.88(1)
RS-10075	-4.78(2)	-4.90(3)	-5.44(3)	-5.81(4)
CFUL658	-4.48(3)	-4.61(3)	-5.10(4)	-5.43(4)
CFUL653	-4.62(2)	-4.78(3)	-5.26(3)	-5.62(3)
CFUF34	-5.09(8)	-5.25(9)	-5.69(18)	-6.30(13)

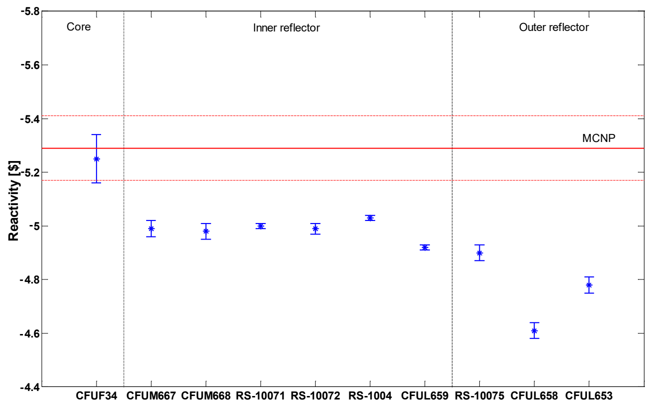

Figure 5. Results for the Sjöstrand method - CR@479.3 mm

This so called spatial effects are the result of system behavior varying from the point kinetic model on which the method is based. In ADS the neutron flux spatial distribution is more uneven than in critical reactors and this flux distribution is not constant. Rather it changes over time after the neutron pulse from the source while it gradually spreads. This gradual flattening of flux distribution is illustrated in the Fig. 6 showing neutron flux calculated with MCNP code for fuel assemblies along core cross section. Spatial effects and their influence on measured reactivity values was observed before in other subcritical systems, such as YALINA [8].

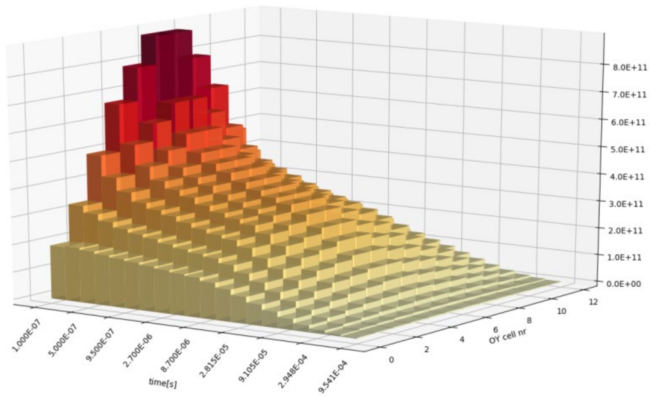

Figure 6. Neutron flux evolution calculated for fuel asseblies along core cross section (neutron flux in arbitrary units) - CR@479.3 mm

IV. Correction Factors

Next step was therefore using a proper correction method that can compensate for the spatial effects. One possible method is to calculate correction factors for each detector position. with the MCNP simulation code. It can be done by calculating core reactivity with KCODE function and the neutron flux for each every detector with and without delayed neutrons. Calculated neutron flux can be then used to calculate expected reactivity value measured in this position [9]:

$$\rho_{SYM} = -\frac{A_P}{A_d} = -\frac{A_{tot}-A_d}{A_d} \qquad (2)$$

Where A_p is neutron flux corresponding to prompt neutrons, A_d is neutron flux corresponding to delayed neutrons (calculated with MCNP) and A_{tot} is total neutron flux (calculated with MCNP).

Correction factor is then calculated as:

$$CF = \frac{\rho_{SYM}}{\rho_{KCODE}} \qquad (3)$$

Corrected reactivity value will be the given by:

$$\rho_{KCODE} = \frac{\rho_{EXP}}{CF} \qquad (4)$$

Where ρ_{EXP} is reactivity value obtained directly from experiment.

Correction factors were calculated separately for each control rod positions. Results are shown in Table II. It has to be noticed that their values do not significantly change with core reactivity, at least in considered range of reactivity values.

TABLE II. CORRECTION FACTORS FOR THE SJÖSTRAND METHOD

Detector	ρ [$]			
	CR@600mm	*CR@479.3mm*	*CR@240mm*	*CR@0mm*
CFUF34	1.001(8)	1.003(7)	0.998(8)	0.994(7)
CFUM667	0.948(7)	0.951(7)	0.946(7)	0.951(8)
CFUM668	0.942(7)	0.949(8)	0.945(8)	0.940(7)
RS-10071	0.937(7)	0.942(8)	0.939(8)	0.946(7)
RS-10072	0.936(8)	0.941(7)	0.937(7)	0.940(8)
RS-10074	0.951(7)	0.947(7)	0.953(7)	0.946(8)
CFUL659	0.946(8)	0.948(9)	0.943(7)	0.940(9)
RS-10075	0.949(9)	0.952(9)	0.947(7)	0.948(8)
CFUL658	0.938(9)	0.941(8)	0.936(9)	0.939(9)
CFUL653	0.936(10)	0.940(9)	0.935(10)	0.939(10)
CFUF34	1.001(8)	1.003(7)	0.998(8)	0.994(7)

The results after applying corrections are shown in Table III and Fig. 7. They show good agreement both with reference values and between different detectors for most of the detectors, with exception to those placed in the outer reflector. However, as it was stated before, the full prompt neutron flux decay wasn't observed in this detectors. Therefore the Sjöstrand method cannot be accurately applied to those detectors. Using lower frequency of the pulses to allow for full prompt decay should allow to obtain proper result. All other detectors show consistent values proving suitability of such correction method. Comparable results were reached by other groups involved in the projects [10, 11].

TABLE III. REACTIVITY VALUES OBTAINED WITH THE SJÖSTRAND METHOD AFTER CORRECTIONS

Detector	ρ [$]			
	CR@600mm	***CR@479.3mm***	***CR@240mm***	***CR@0mm***
CFUF34	-5.08(9)	-5.23(10)	-5.70(19)	-6.34(14)
CFUM667	-5.12(5)	-5.25(5)	-6.12(13)	-6.33(9)
CFUM668	-5.17(5)	-5.25(5)	-5.97(12)	-6.36(9)
RS-10071	-5.13(5)	-5.31(5)	-5.91(5)	-6.30(6)
RS-10072	-5.14(5)	-5.30(5)	-5.99(6)	-6.33(6)
RS-10074	-5.09(4)	-5.31(5)	-5.90(6)	-6.32(6)
CFUL659	-5.02(4)	-5.19(5)	-5.83(5)	-6.26(5)
RS-10075	-5.04(5)	-5.15(5)	-5.74(6)	-6.13(7)
CFUL658	-4.78(5)	-4.90(5)	-5.45(6)	-5.78(7)
CFUL653	-4.94(5)	-5.09(5)	-5.63(6)	-5.99(6)
CFUF34	-5.08(9)	-5.23(10)	-5.70(19)	-6.34(14)

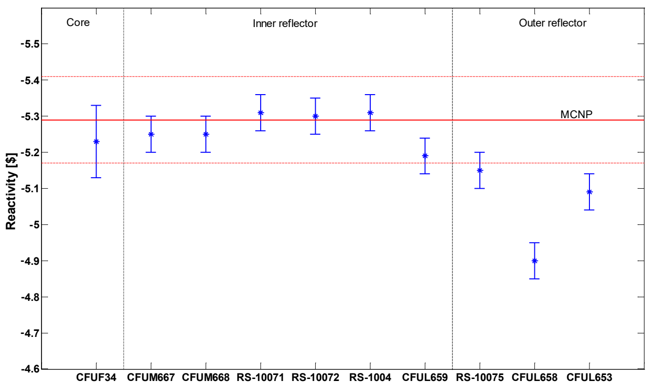

Figure 7. Results for the Sjöstrand method after corrections - CR@479.3 mm

V. SUMMARY AND CONCLUSIONS

It was shown, both the experiment and numerical modeling, that core kinetic behavior of an ADS differs from point kinetic model. It affects reactivity measurements done by using methods derived from this model. This so called spatial effects means that measured reactivity value depends on detector placement within the system. It is consistent with previous experimental studies on ADS reactivity measurement [8, 9] despite different neutron spectra.

It was again confirmed that calculating proper correction factors can be used to correct for spatial effects in PNS measurement and allows to obtain correct reactivity value regardless of detector placement. It has to be however noticed that use of this method in real-time measurement would require calculating the correction factors in advance. A study of robustness of this method should be therefore performed to asses if those correction factors change with reactivity and other core conditions. It would be possible to use several sets of precalculated correction factors, but an assessment of range of conditions that once calculated correction factor can be used is required. It should take into account reactivity itself, neutron flux distribution, fuel loading patterns and burnup. It should be however noticed that in the range of reactivity (about 1.5$) values covered by the experiment the correction factors did not change. Neither would they be significantly affected by uneven insertion of the control rods in such core configuration and resulting uneven neutron flux distribution [12]. Therefore it may be concluded that change of the correction factors would be slow enough that use of this method would be practical.

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IYNC2020 SYDNEY

Digital engineering design platform data analysis and machine learning algorithm

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I. INTRODUCTION

In the global nuclear power plant construction market, a mistake in a project can cost a company its reputation. Today, Rosatom is constructing 42 power units in 12 countries, and another 30 facilities are at the stage of contracting [1]. Each project has its own regulator who imposes strict requirements on the project and monitors compliance with these requirements.

In a highly competitive market, we need to respond quickly to new customer requirements for the content and execution of the project documentation and promptly inform each specialist who works on the project. It is also important to keep in mind our experience and to retain critical knowledge.

II. ABOUT THE PROBLEM

Every year, each design engineer in the ASE group develops more than 125 A1 drawings containing complex technical solutions and this amount is increasing every day. Under tight deadlines, work on several projects with different requirements is carried out simultaneously. Under these conditions, the probability of design errors is especially high.

Each project error is the reason that the customer does not accept the project documentation from the first time and returns it to the developer for revision. Sometimes these errors are insignificant and arise due to the loss of attention in the development and lack of effective communications. At the same time, each of them has a negative impact on the company reputation. The elimination of errors leads to additional labor costs and payment of works to correct the project, as well as increases the likelihood of failure to meet construction deadlines.

To avoid these risks, each developed drawing and individual document should be checked by the developer and his supervisor. Verification is performed visually for compliance of documents with certain criteria fixed in the checklist. Criteria are formed on the basis of normative documents, requirements of regulators for project documentation and experience of interaction with it. This approach is widely used in most engineering companies. At the same time, this approach is conservative and requires additional human resources.

The solution to this problem is in the field of automation of project documentation verification using big data analysis and machine learning technologies [2]. A distinctive feature of machine learning methods is their focus on working with data: most often they work with a certain sample of data, extract patterns from it and build some predictions on new, previously unobserved data [3]. Therefore, the term "data analysis" is also used to refer to this area of knowledge.

III. RESULTS

As a solution, our team of young specialists offers to create a digital platform for automated verification of project documentation compliance with customer requirements, regulations and requirements for the design of the project, information about errors in the project documentation, as well as for generation of personal recommendations to users to prevent them. The idea is based on a machine learning algorithm that can recognize graphical and textual information and correctly interpret it by semantic analysis.

The algorithm of the digital platform is built as follows:

1. The user uploads the project documentation to be checked to the platform. The download is done in an Internet browser, which does not require a large PC performance, as the platform is built on the basis of cloud technologies.
2. Using machine vision technology, the system recognizes and interprets the data in the processed documents. The platform is able to recognize text documents, tables with data, as well as graphical information (drawings, diagrams).
3. The system compares the recognized data with the reference data for a certain project. Comparison is performed automatically according to pre-configured verification algorithms for the project. Each document undergoes a comprehensive review. The contents of the document is tested against the criteria, the document is validated against related documents in the project, and in terms of identity of data within them and their compliance with the design documentation structure.
4. The results of the comparison are formed for the user as a list of revealed inconsistencies. In this case, the user has the opportunity to make additional notes in the generated report for

further work. The information is automatically stored in the cloud storage.

5. Simultaneously with the data verification, the platform collects and accumulates information about inconsistencies, and then, based on the methods of machine learning, forms personal recommendations to the user to prevent the most common errors. The platform also accumulates information on errors in the project documentation, on the basis of which we can improve our efficiency.

The prototype of the platform user interface is shown in figure 1.

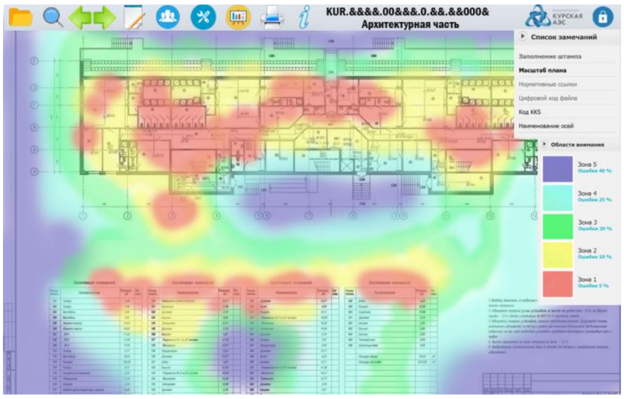

Figure 1. The prototype of the platform user interface.

The project team conducted a comparative analysis of existing solutions in the field of automation of project documentation verification and our proposed solution. The results of the analysis are shown in table 1.

TABLE I. COMPARISON OF THE PRODUCT WITH ANALOGUES

Feature	**Digital product**			
	Our solution	***AUTODESK Design Review***	***Solidworks 3D automatic drawing checker***	***Visual Engineer***
Design requirements control	●	●	●	●
Digital marks	●	●		
Machine learning	●			
Requirements change management	●		●	
Reports	●	●		●
Personal recommendations	●			

At the moment, according to our technical task, a prototype of the digital platform is being developed. For this purpose, we have analyzed more than two thousand sets of project documentation (about 20% of the total volume of NPP project documentation). The collection and analysis of project data continues in order to improve the platform validation and training algorithms.

According to our forecast, the use of the platform will reduce the number of potential errors in the project documentation by 34 %. Design engineers and their supervisors will save up to 17% of the time they previously spent checking project documentation. In general, these measures will improve the quality of documentation and customer satisfaction from cooperation with our company. Our project is the next step of Rosatom towards digitalization in design and knowledge management.

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Preliminary assessment of decay heat removal systems in the ESFR concept: the role of natural air convection around Steam Generators outer shells in accidental conditions

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I. INTRODUCTION

A new European project, entitled ESFR-SMART, was launched in 2017 for a 4-year period [1], [2]. This project's goal is to improve the safety level of the European Sodium cooled Fast Reactor (ESFR) by simplifying the design and using positive features of Sodium cooled Fast Reactors (SFR). The ESFR concept embeds, as presented in Fig. 1, a sodium pool type vessel containing the reactor core and is associated to 6 Intermediate Heat Exchangers (IHX). Each IHX is linked to 6 modular Steam Generators (SGs) which are located in casings. In case of an accidental transient with loss of electrical supply, it is possible to evacuate a part of the decay heat generated from the core via these casings, by cooling the SGs outer shells by natural convection of air thanks to openings located at the bottom and the top of each casing (system called DHRS-2). This paper describes the calculations performed to assess the capability of the SGs to evacuate the decay heat from the reactor vessel. Both theoretical and CATHARE code (Thermal Hydraulics reference code) [3] calculations are presented. The impact of both an additional chimney at the top of each casing as well as running primary and secondary pumps on the heat removal capacity are evaluated.

II. THEORETICAL EVALUATIONS

A. Methodology

In order to evaluate the amount of decay heat generated from the core that can be extracted from the reactor vessel thanks to natural circulation via the SGs casings, different energy balance equations are established in steady state in each system: the reactor vessel, the intermediate loops, and the SGs casings. This theoretical analysis is performed in addition to the CATHARE evaluation to make sure CATHARE results are in accordance with a simplified theoretical model. Modeling natural circulation around SGs casings with CATHARE is complex and provide theoretical solutions is relevant.

B. Energy balance in the reactor vessel

The energy balance between the core decay heat (Q), the total natural circulation flow $W_{Na\,Vessel}$ in the vessel, the specific heat capacity of the sodium in the reactor vessel $Cp_{Na\,Vessel}$ and the temperature at the inlet and outlet of the core (respectively T_{Ves}^{Cold} and T_{Ves}^{Hot}) is presented in (1):

$$Q = W_{Na\,Vessel}.Cp_{Na\,Vessel}.(T_{Ves}^{Hot} - T_{Ves}^{Cold}) \quad (1)$$

The thermal pump ΔP produced by the core and the heat exchangers can be evaluated as follows:

$$\Delta P = \Delta\rho.g.H \quad (2)$$

Where $\Delta\rho$ is the sodium density variation associated to the sodium heating, and H the height of heat exchangers and the core.

The head losses in the reactor vessel (ΔP) are directly linked to the natural circulation flow $W_{Na\,Vessel}$ in the vessel, as presented in equation (3). The k_{vessel} parameter is the global head loss coefficient in the reactor vessel:

$$\Delta P = k_{vessel}.W_{Na\,Vessel}^{\ 2} \quad (3)$$

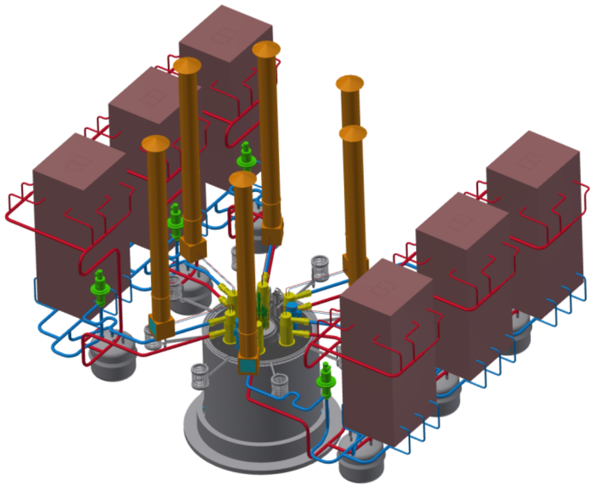

Figure 1. ESFR-SMART main view of the reactor without chimneys above the casings of the steam generators [1]

The density variation of sodium ($\Delta\rho$) is related to the core inlet and outlet temperatures thanks to equation (4):

$$\Delta\rho = \beta_{Na}.(T_{Ves}^{Hot} - T_{Ves}^{Cold}) \quad (4)$$

ESFR-SMART European Project

In equation (4), β_{Na} is the linear coefficient linking density variation to temperature variation. A polynomial equation linking sodium density variation to temperature could be used but would only increase the precision of the calculation of a few percent, which is not relevant for this preliminary theoretical analysis. In steady state, the pressure increase caused by the core and the heat exchangers (2) exactly compensates the head losses in the vessel (3). Combining equations (2), (3) and (4), equation (5) can be written:

$$\Delta P = \Delta\rho.g.H = \beta_{Na}.(T_{Ves}^{Hot} - T_{Ves}^{Cold}).g.H = k_{vessel}.W_{Na\,Vessel}{}^2 \quad (5)$$

Hence:

$$W_{Na\,Vessel} = \gamma_{Vessel}\sqrt{(\mathrm{T}_{Ves}^{Hot} - \mathrm{T}_{Ves}^{Cold})} \quad (6)$$

With: $$\gamma_{Vessel} = \sqrt{\frac{\beta_{Na}.g.H}{k_{vessel}}} \quad (7)$$

Using the Log Mean Temperature Difference (LMTD), it is possible to write for each IHX equation (8):

$$\frac{Q}{6} = K_{IHX}.S_{IHX}.\left[\frac{(T_{Ves}^{Hot} - T_{IHX}^{Hot}) - (T_{Ves}^{Cold} - T_{IHX}^{Cold})}{ln\left(\frac{T_{Ves}^{Hot} - T_{IHX}^{Hot}}{T_{Ves}^{Cold} - T_{IHX}^{Cold}}\right)}\right] \quad (8)$$

K_{IHX} is the exchange coefficient for each IHX ($\mathrm{W.m^{-2}.K^{-1}}$) and S_{IHX} the exchange surface area for each of the 6 IHX ($\mathrm{m^2}$). T_{IHX}^{Hot} and T_{IHX}^{Cold} are the temperatures of the secondary side sodium fluid just before and after the exchange with IHX.

C. Energy balance in the SGs casings

The same methodology as presented in II.B applies to the SGs casings and results in equations (9), (10), (11) and (12):

$$\frac{Q}{6} = W_{Air}.Cp_{air}.(\mathrm{T}_{Air}^{Hot} - \mathrm{T}_{Air}^{Cold}) \quad (9)$$

W_{air} is the air flow in each SG casing, Cp_{air} is the specific heat capacity of the air and T_{Air}^{Cold} and T_{Air}^{Hot} are respectively the temperatures of air entering and exiting each SGs casings.

$$W_{Air} = \gamma_{SGs\,Casing}\sqrt{(\mathrm{T}_{Air}^{Hot} - \mathrm{T}_{Air}^{Cold})} \quad (10)$$

$$\gamma_{SGs\,Casing} = \sqrt{\frac{\beta_{Air}.g.H_{SG}}{k_{Casing}}} \quad (11)$$

In equation (11), β_{Air} is the linear coefficient linking density variation to temperature variation. LMTD leads to:

$$\frac{Q}{6} = K_{Cas}.S_{Cas}.\left[\frac{(\mathrm{T}_{IHX}^{Hot} - \mathrm{T}_{Air}^{Hot}) - (\mathrm{T}_{IHX}^{Cold} - \mathrm{T}_{Air}^{Cold})}{ln\left(\frac{\mathrm{T}_{IHX}^{Hot} - \mathrm{T}_{Air}^{Hot}}{\mathrm{T}_{IHX}^{Cold} - \mathrm{T}_{Air}^{Cold}}\right)}\right] \quad (12)$$

K_{Cas} is the exchange coefficient for each casing ($\mathrm{W.m^{-2}.K^{-1}}$). This coefficient is evaluated in §II.E. S_{Cas} is the exchange surface area for each SGs casing ($\mathrm{m^2}$). As mentioned above, each SGs casing contains 6 SGs. Hence, S_{Cas} is the exchange surface area for 6 SGs. T_{IHX}^{Hot} and T_{IHX}^{Cold} are defined in II.B. It is supposed that the heat losses in the secondary circuit are negligible.

D. Energy balance in the secondary side loops

In the same way as for the reactor vessel and the SGs casings, it is possible to write 2 more equations (13 and 14) for each secondary side loop:

$$\frac{Q}{6} = W_{Na\,Sec}.Cp_{Na\,Sec}.(\mathrm{T}_{IHX}^{Hot} - \mathrm{T}_{IHX}^{Cold}) \quad (13)$$

$$W_{Na\,Sec} = \gamma_{Na\,Sec}\sqrt{(\mathrm{T}_{IHX}^{Hot} - \mathrm{T}_{IHX}^{Cold})} \quad (14)$$

$W_{Na\,Sec}$ is the flow of sodium in each secondary side loop and $Cp_{Na\,Sec}$ is the specific heat capacity.

E. Exchange coefficients evaluation

Both IHX and SGs exchange coefficients, respectively K_{IHX} and K_{Cas} need to be evaluated. This is done using the Dittus-Boelter correlation [4] for the IHX (natural circulation in pipes) and Morgan empirical natural convection correlations [5] for the SGs casings. Nominal values are 874.5 $\mathrm{W.m^{-2}.K^{-1}}$ for the IHX and 15.9 $\mathrm{W.m^{-2}.K^{-1}}$ for the SGs casings.

F. Equation solving

By combining equations (1) to (14) and assuming that the temperature of air entering the SGs casings, $\mathrm{T}_{Air}^{Cold}$is known, all the key parameters can be calculated. The most relevant parameter is the sodium temperature getting out of the core. This temperature should not exceed 650°C to protect the vessel mechanical integrity (decoupling value).

Defining φ_1 that is a known coefficient in equation (15); it is possible to calculate T_{IHX}^{Hot} thanks to equation (16):

$$\varphi_1 = e^{\frac{6}{Q}K_{Cas}.S_{Cas}.\left(\left(\frac{Q}{6\gamma_{Na\,Sec}.Cp_{Na\,Sec}}\right)^{\frac{2}{3}} - \left(\frac{Q}{6\gamma_{SGs\,Casing}.Cp_{air}}\right)^{\frac{2}{3}}\right)} \quad (15)$$

$$T_{IHX}^{Hot} = T_{Air}^{Cold} + \frac{1}{1-\varphi_1}\left[\left(\frac{Q}{6\gamma_{SGs\,Casing}.Cp_{air}}\right)^{\frac{2}{3}} + \varphi_1\left(\frac{Q}{6\gamma_{Na\,Sec}.Cp_{Na\,Sec}}\right)^{\frac{2}{3}}\right] \quad (16)$$

G. Results

For nominal values of exchange coefficients, theoretical analysis predicts that around 17 to 18 MWth can be evacuated by circulation of air around the SGs in the casings (for a T_{Ves}^{Hot} equaled to 650°C), corresponding to the decay heat generated by the core around 48 hours after the reactor shutdown. Table I presents the power evacuated by the SGs casings for different exchange coefficients values (increased or decreased of 20% compared to nominal values) at a fixed cold air temperature of 20°C. This shows that IHX exchange coefficient is not a sensitive parameter since the exchange surface is high (about 2000 m²) and exchanged power relatively low compared to nominal power exchanged (3600 MWth). The exchange coefficient for the SGs casings appears to play a significant role regarding the power evacuated.

TABLE I. POWER EVACUATED BY STEAM GENERATORS

T_{Air}^{Cold}(°C)	*20*	*20*	*20*	*20*	*20*
T_{Air}^{Hot}(°C)	*144.5*	*158.9*	*144.7*	*131.6*	*144.7*
K_{IHX} $(W.m^{-2}.K^{-1})$	*874.5*	*874.5*	***1050***	*874.5*	***729***
K_{Cas} $(W.m^{-2}.K^{-1})$	*15.9*	***19.1***	*15.9*	***13.2***	*15.9*
T_{Ves}^{Hot}(°C)	650.0	650.0	650.0	650.0	650.0
Power evacuated (MWth)	*17.55*	*20.63*	*17.56*	*14.86*	*17.55*

III. CATHARE CALCULATION

The CATHARE code (Code for Analysis of THermalhydraulics during an Accident of Reactor and safety Evaluation) has been used to evaluate the power that can be evacuated by the circulation of air around the SGs outskirts in the casings during a Loss Of Offsite Power (LOOP) transient. It is assumed to happen at the initial time while the reactor is under nominal operating conditions (3600 MWth produced in the core). The LOOP leads to the reactor scram as well as the loss of all active systems (all pumps): decay heat is only removed by the air circulation around the SGs after they have dried out (shortly after the beginning of the accident).

A. Methodology

A CATHARE input deck has been created to model ESFR-SMART reactor design. In order to have the most reliable modeling all three circuits have been considered in this deck: the reactor vessel, the secondary loops and the SGs located inside the casings. The most challenging part of creating this input deck is related to the SGs and the modeling of air circulation around them. To model the heat exchange between sodium inside SGs and the air inside the casings in contact with the outer part, a thermal wall has been added to the SGs. This enables to model this exchange through a heat exchange coefficient (parameter HEXT) and the air temperature inside the casing (parameter TEXT).

The air temperature is considered to be constant due to the large volume inside each of the casings, which is a difference between theoretical and CATHARE calculations. CATHARE calculation is hence a little bit less conservative than theoretical evaluations. The airflow is not directly modeled by CATHARE, but depends on the heat exchange efficiency, so that the air flowrate may be modeled implicitly through the value of CATHARE parameter HEXT.

B. Results

Fig. 2 shows the evolution of the core inlet and outlet temperatures inside the reactor vessel over 24 hours, for a fixed value of HEXT equaled to 20 $W.m^{-2}.K^{-1}$ and a temperature of cold air of 20°C. CATHARE calculation is stopped after around 24 hours of transient because of numerical instabilities. Fig. 2 shows that after around 7 hours of transient, the temperature of the sodium out of the core exceeds the 650°C criterion. The power evacuated by the SGs, presented in Fig. 3, is not sufficient to prevent high sodium temperatures in the vessel. One way to increase the power evacuated is to add chimneys (cf. IV.A).

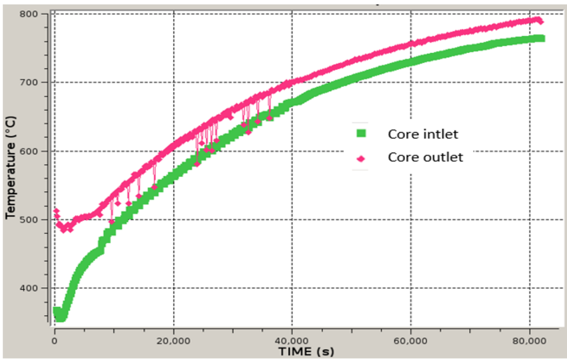

Figure 2. Core inlet and outlet temperatures during the LOOP transient

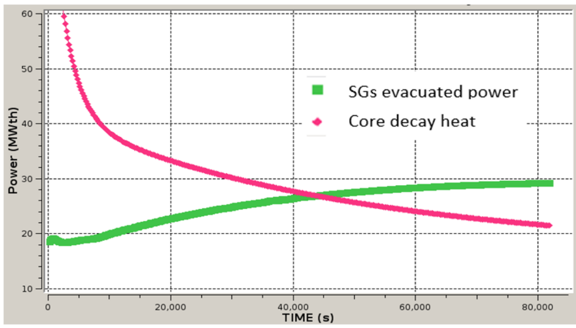

Figure 3. Core decay heat and power evacuated by SGs

Fig. 3 shows that even if the core decay heat (evaluated by CATHARE using eight energy groups) is lower than the power evacuated by the SGs, the temperature in the reactor vessel is increasing. This is explained by the enormous inertia of the reactor vessel and secondary circuits and the small flows of sodium in the vessel and different loops during natural circulation. After one day of transient, the decay heat is around 20 MWth [1]. Fig. 2 and 3 also show that, for example, for a temperature at the outlet of the core of around 800°C, the power evacuated by the SGs is around 29 MWth. Exactly the same result is obtained when the theoretical equations (described in §II) are applied. This validates that the theoretical correlations enable to estimate correctly the reactor configuration (temperatures in the vessel, power extracted by the SGs…) compared to CATHARE reference code.

IV. SENSITIVITY ANALYSES

A. Impact of adding chimneys

Since the natural convection of air in the SGs casings does not prevent from reaching high temperatures in the reactor vessel, the impact of adding one chimney (Fig. 4) at the top of each casing to increase the airflow and the evacuated power is evaluated. The depressurization induced by each chimney, $\Delta P_{chimney}$ depends on inlet conditions (hot air density ρ_h) and outlet conditions (ambient air density ρ_0) of the chimney and on its height $h_{chimney}$. It is evaluated thanks to equation (18):

$$\Delta P_{chimney} = (\rho_0 - \rho_h).g.h_{chimney} \quad (18)$$

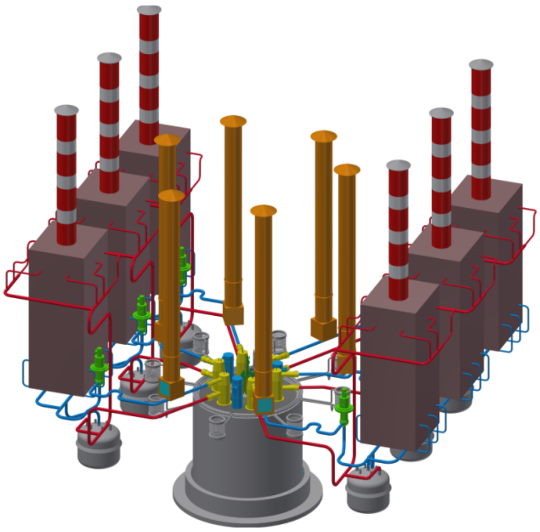

Figure 4. ESFR-SMART main view of the reactor with chimney addition

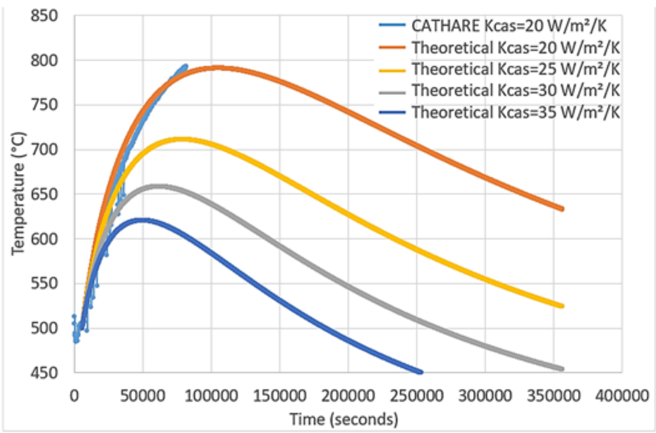

Figure 5. Core outlet temperature (°C) for different values of K_{Cas} vs time (s)

A theoretical analysis (not described in this paper) shows that taking into account chimneys of a height of 45 meters increases the exchange coefficient K_{Cas} of 54% from 15.9 $W.m^{-2}.K^{-1}$ to 24.5 $W.m^{-2}.K^{-1}$. By adjusting the reactor inertia (I) to fit to CATHARE sodium temperature increase in the reactor vessel and applying equation (19), it is possible to show (Fig. 5) that for a K_{Cas} of 24.5 $W.m^{-2}.K^{-1}$, the maximum temperature in the reactor vessel should still exceed 650°C, which is not sufficient to evacuate the decay heat safely.

$$I.\frac{dT}{dt} = Q_{Decay}(t) - Q_{Exchanged\ SGs}(t) \quad (19)$$

B. *Primary and secondarty side pumps running*

An additional evaluation is performed to determine if assuming the primary and secondary side pumps are running, the decay heat can be removed safely by the SGs via natural circulation of air. One chimey is considered to be at the top of the each casing. The same methodology based on the CATHARE result extrapolation is applied (§IV.A). Fig.6 shows that in this configuration, the maximum core outlet temperature during the LOOP transient should be of 620°C and meet the safety criterion.

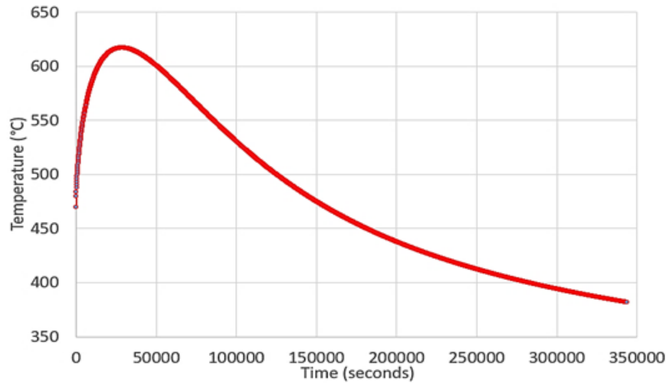

Figure 6. Core outlet temperature (°C) for running pumps vs time (s)

V. CONCLUSIONS

This paper shows that it is possible to evaluate the power evacuated thanks to the natural circulation of air around SGs using theoretical equations that match well compared to CATHARE results. In a “Fukushima situation” (with no water in the SGs and no power supply), the natural circulation of sodium cannot safely remove alone the decay heat generated by the core since the temperature of the sodium in the reactor vessel temporarily exceeds the safety criterion of 650°C. The addition of chimneys increase the capacities but is not sufficient to evacuate the decay heat safely. If the primary and secondary side pumps are running, the safety criterion should be met.

ACKNOWLEDGMENT

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Thermal-hydraulics of loss of heat sink accident of Indian test blanket system in ITER

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I. Introduction

The two isotopes of hydrogen (Tritium and deuterium) will be used as fuel in fusion reaction in future coming fusion reactors. Since the limited supply of tritium is available in the world, we have to generate it artificially within the tokamak by the interaction of neutrons escaping from the plasma with lithium which is contained in the blanket. The world's largest experimental fusion project (ITER) will provide a unique opportunity to test mockups of breeding blankets also called Test Blanket Modules (TBM) of each ITER participating country to collect DEMO relevant experimental data and for that ITER has provided dedicated ports in the machine. The key feature of TBM is to develop the design technology for DEMO and future power-producing fusion reactor. India has proposed LLCB TBM as the blanket concept. The TBM design should have sufficient features to capture any unwanted accident within the design limits prescribed by ITER and before testing each country should have to perform thermal-hydraulic safety analysis of its test blanket system (TBS). The objective of the safety analysis is to demonstrate that, the Test Blanket System (TBS) design has sufficient provisions to withstand accident sequences without violating the release guidelines and other safety principles established for ITER and documented in the Safety Guidelines for Test Blanket Systems [1]. Under this exercise, we have chosen a loss of heat sink accident (failure of heat exchanger secondary side pump), a potent accident that may affect ITER operation. The other potent reference accidents were already analyzed in articles [1,2,3 and 4].

II System Description and Components

Indian lithium lead ceramic breeder test blanket system (LLCB TBM) has two main cooling systems: First Wall Helium Cooling System (FWHCS) and Lead Lithium Cooling System (LLCS). The FWHCS is a high-pressure high-temperature system used for cooling TBM First Wall, top plate, bottom plate, and backplate. The FWHCS is located in the tokamak cooling water system (TCWS) vault annex and it supplies helium to the inlet of the TBM. The helium outlet from TBM is routed back to the TCWS. Depending on the tritium concentration in circulating helium the bleed line is opened to coolant purification system for purification and recirculation of helium gas. The total heat removed by this loop under normal condition is about 0.3 MW. The LLCS provides the required flow of liquid lead-lithium for cooling the TBM. The lead-lithium flows through the internal channels surrounding ceramic breeder compartments of TBM. The LLCS system consists of a dump tank, mechanical pump, lead-lithium to the helium heat exchanger and detritiation system, all located in the port cell. Trace heaters keep the complete loop at high temperature to avoid freezing of Pb-Li. In the current exercise, we have performed only the thermal-hydraulic safety analysis of the FWHCS loop. The Indian LLCB blanket concept consists of lithium titanate as a ceramic breeder (CB) material in the form of packed pebble beds, Lead-Lithium (Pb-Li) is acting as a coolant for ceramic breeder beds and in addition acts as tritium breeder and multiplier. The high-pressure helium gas cools the box structure of the blanket module such as the First wall, top plate, bottom plate, and backside plate [2]. The First Wall Helium Cooling System (FWHCS) transports the heat from the FW and the outer box structure. The TBM first wall is cooled by high-pressure primary helium, which rejects heat to ITER water cooling system. The FWHCS is designed to remove the peak heat load of 300 kW [3]. The block diagram of FWHCS of LLCB TBM is documented in reference [3]. The TBM FW composed of a 28 mm thick U-shaped RAFMS structure, having internal cooling channels of 20 mm × 20 mm cross-section. The coolant channels are designed to allow multiple passes of helium coolant across the FW in order to maximize the heat removal. The number of helium passes has been optimized such that the maximum temperature in the RAFMS remains below the design limit of 550◦C. The FW structure is having 64 helium coolant channels [4]. The schematic of the TBM system is given in Fig. 1. Figure 2 shows the interaction boundary of the FWHCS and LLCS loop. The thermal-hydraulic nodalization of the FWHCS is shown in Fig. 3. The simulation of the accident is performed by thermal-hydraulic safety analysis code RELAP/SCADAPSIM/ MOD4.0.

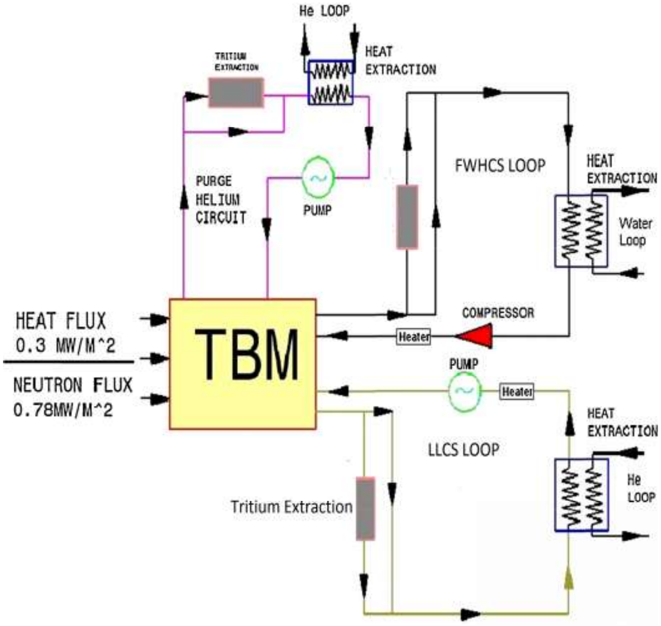

Figure 1. Schematic of the TBM cooling systems

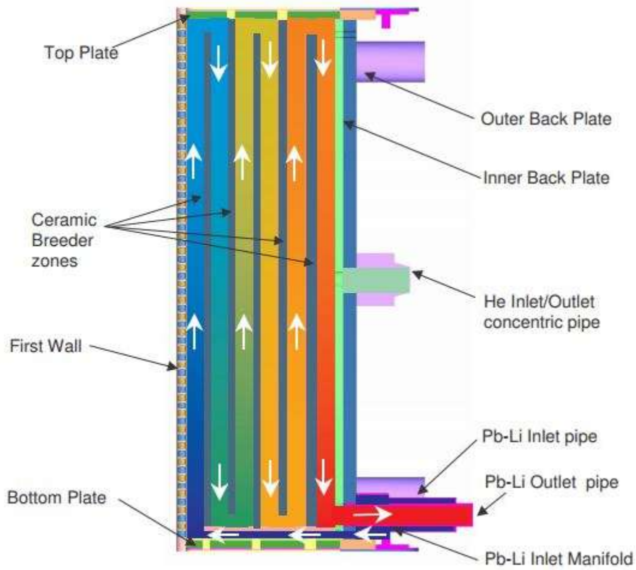

Figure 2. FWHCS and LLCS interaction boundry

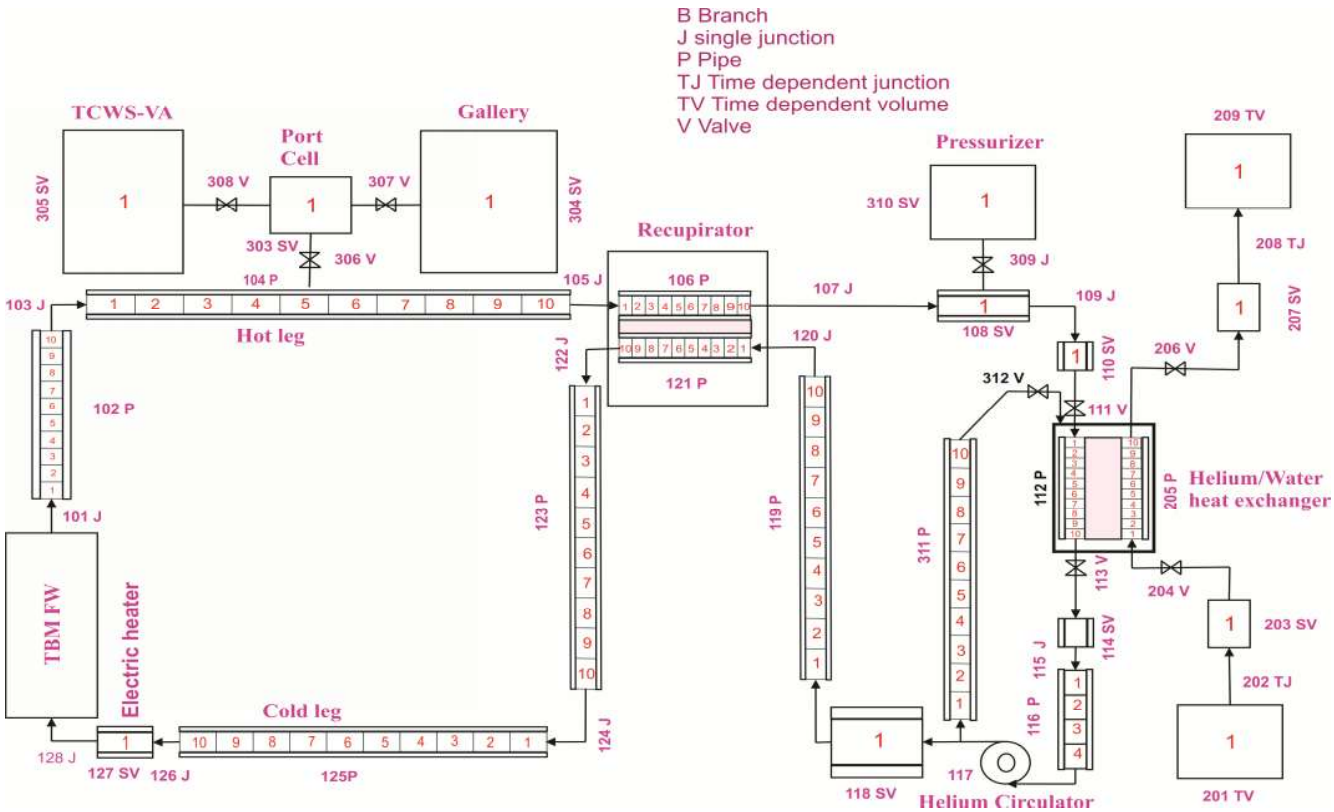

Figure 3. Thermal hydraulic nodalization of first wall helium cooling system

III Results and analysis

The postulate initiating event (PIE) of this accident is the trip of secondary water side centrifugal pump of the FWHCS heat exchanger. Due to the decrease of mass flow rate in the secondary water side of the heat exchanger, the heat removal capability of FWHCS is reduced. As the mass flow rate reduced to 70 % of its steady-state value, the fast plasma shutdown system (FPSS), trips the reactor by the detection of low flow signals from the system. The accident is assumed to start at the end of 500 MW plasma pulse and a 20% power excursion for 10 s before PIE is also assumed to guarantee peak TBM first wall temperatures during the time span of the accident. Due to lack of coolability of secondary side, less amount of heat is removed from the primary helium side which will lead to melting of first wall channels and high pressure (0.8 MPa) helium gas ingress into vacuum vessel cause plasma induce intense plasma disruption and deposits plasma stored thermal energy of 1.8 MJ/m^2 over a period of time assumed to be 1 s in duration. Multiple TBM FW cooling tube failure within a 0.1 m high toroidal strip [5] is assumed. The total size of the break is defined as the double-ended rupture of all coolant channels within the toroidal strip around the TBM and this represents 4 FW channels (total break size $32x10^{-6}$ m^2).

The transient starts with a secondary side pump trip at the end of the flat top of a 500 MW pulse. Fig. 4 shows the mass flow rate at the inlet of TBM. After 2 s of the accident, the mass flow rate in the secondary loop reduces to 70 % and FPSS trips the reactor. Fig. 5 shows the temperature evolution during the loss of heat sink accident with and without FPSS trip activation. The FW temperature increases rapidly without FPSS activation because of continuous heat deposition from the plasma which will subsequently lead to failure of first wall cooling channels. In case of loss of heat sink accident with FPSS signal, the temperature initially increases followed by disruption heat with subsequent slow decrease profile due to plasma shut down (less amount of decay heat and heat removed by radiation). As shown in Fig. 5 and 6, a peak temperature of 598°C is observed at 3.1 s. Fig. 6 shows the TBM FW temperature evolution during the accident and during 10 days (with decay heat). Because of the 20 % power excursion for 10 s before first wall leak [6], average first wall temperature rise from 450°C to 475°C and thereafter a peak temperature of ~598°C is observed after 1 s of the accident, due to the deposition of plasma disruption heat load (1.8 MJ for one second).

The temperature initially reduces with a fast rate due to the high helium mass flow rate coming out from the first wall break. After few seconds of the accident the first wall cooling is completely lost and then after remaining passive heat is removed by radiation loss and conduction to the colder structure which causes a slow decrease in temperature with time as shown in Fig. 6, the temperature reduced to 253°C after 10 days of the accident. Fig. 7 shows the pressure profile of the TBM first wall and vacuum vessel (VV) during the accident. The FW helium pressure of 8 MPa is rapid decreases because of loss of helium into vacuum vessel and 15 s after the first wall break pressure equalizes to VV pressure.

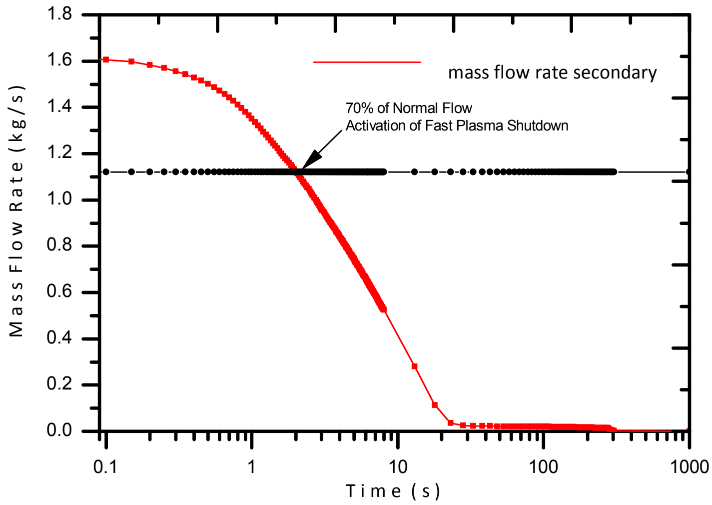

Figure 4. Mass flow rate secondary water side during LOHS accident

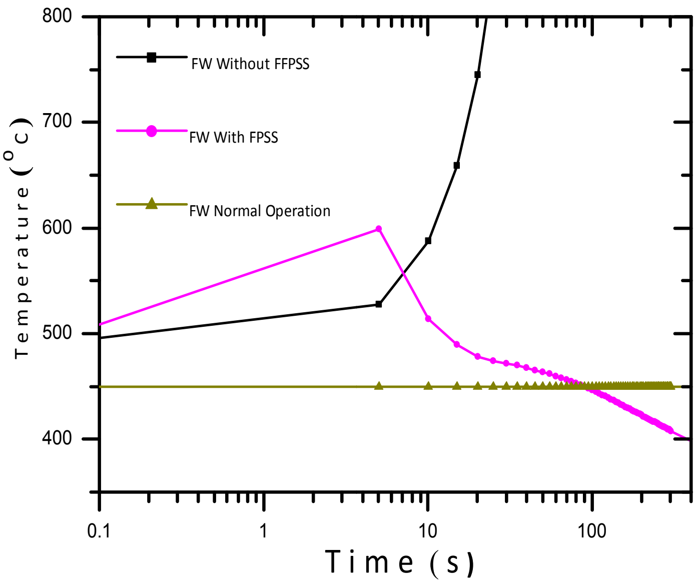

Figure 5. TBM FW temperature evolution with and without FPSS during LOHS accident

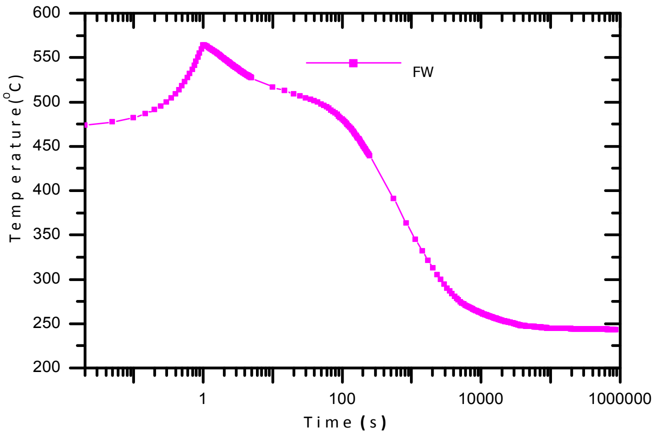

Figure 6. TBM FW temperature evolution for 10 days during LOHS accident (With decay heat)

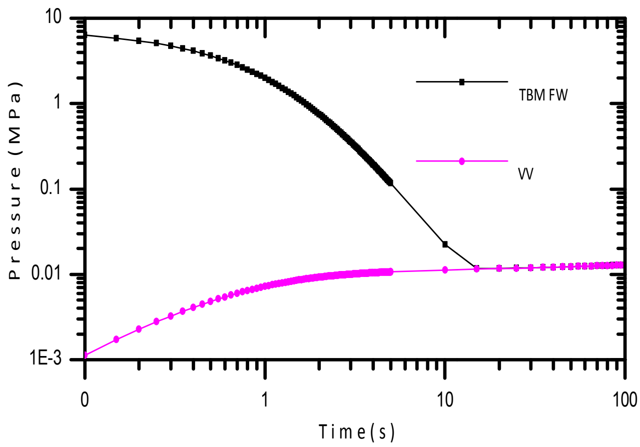

Figure 7. Mass flow rate heat exchanger secondary water side during LOHS accident

CONCLUSIONS

The analysis shows that in case of loss of heat sink accident the important parameters like first wall temperature, vacuum vessel pressure are under the prescribed safety limit provided by the ITER organization. The important findings are as follows: The total helium inventory of the FWHCS loop does not cause over-pressurization of VV and VV pressure remains well below the design limit (0.2 MPa). The maximum first wall surface temperature is about 598^{o}C. The analysis also shows the passive heat removal capability of the TBM structure. Analysis of LOFA in FWHCS shows that activation of FPSS following the event is necessary in order to prevent TBM FW failure.

APPENDIX

NOMENCLATURE

DEMO	Demonstration nuclear fusion power plant
FW	First Wall
FWHCS	First wall helium cooling system
FPSS	Fast Plasma Shutdown System
ITER	"The Way" in Latin
LOHS	Loss of Heat Sink
LOFA	Loss of flow accident
PIE	Postulated initiating Event
ST	Suppression Tank
TBM/TBS	Test Blanket Module/Test Blanket System
VV	Vacuum Vessel

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Degraded core relocation in Sodium-cooled Fast Reactor severe accident – particle-size debris flow

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I. Introduction

A new generation of nuclear reactors is investigated with the criteria of sustainability, enhanced safety, economics, and proliferation resistance. Among different designs of GEN IV reactors, Sodium-cooled Fast Reactors (SFR) are chosen as reference systems due to the most extensive industrial experience and operational feedback available for this type. Under the research and development of SFR reactors, the domain of severe accident is addressed with high priority in the context of improved safety requirements. In the French frame of SFR safety research, oriented mainly around ASTRID reactor, an innovative severe accident mitigation architecture is being investigated. In this paper, the safety study approach and the mitigation strategy is introduced.

The evaluation of the mitigation performance is currently based on best-estimate calculations with the reference computer code SIMMER. However, uncertainties of SIMMER approach, closely related to the simulation of mitigation scenario and influencing the predicted performance, have been identified. In the second part of the paper, the uncertain modelling area of our interest is described and the first verification & validation of SIMMER are presented.

II. Severe accident in Sodium-cooled Fast Reactors

The French safety study approach of future generation of SFRs is based on identification of different severe accident families. The families are representative initiating events having the potential to lead to severe core degradation. There are three main categories defined:

- reactivity insertion (unprotected transient overpower, UTOP),
- core cooling failure (unprotected loss of flow, ULOF),
- local cooling fault accidents (unprotected sub-assembly fault, USAF).

Each family of sequences can be decomposed into four accidental phases primary, transition, secondary and post-accidental, Figure 1. [1]. The decomposition permits to better follow transient evolution and to point out the driving phenomena.

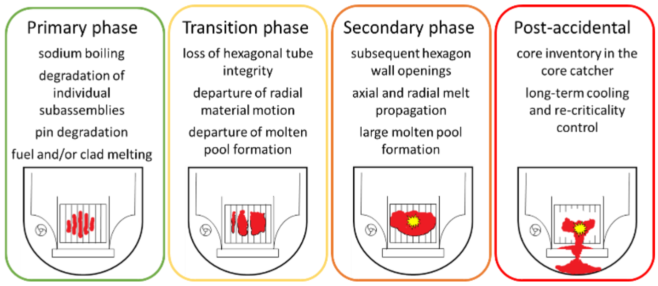

Figure 1. Decomposed accidental phases.

Over the years of design, manufacturing, commissioning, operation and functioning experience, SFR systems have demonstrated several intrinsic advantages but raised also various problematic areas that still lack the knowledge to resolve. The most crucial aspect regarding the debated safety issue is related to the reactivity characteristic of the sodium-cooled core. Fast reactor cores are not in their most reactive configuration therefore any compaction inserts positive reactivity and enhances the possibility of reaching prompt criticality. Compaction takes place in the transition and secondary phase, when the internal geometry is modified and the relocation of melted core materials is such that dense reactive molten pools may be formed. Prompt criticality induces rapid heating and high pressurization, which disassemble core structures and vaporize fuel and steel components. The resulting large mechanical energy release has the potential to damage reactor vessel and challenges containment integrity. For this reason, core disruptive accidents are treated as major issue in SFR severe accident management .

The purpose of severe accident management is to ensure the safe termination of a postulated core degradation event without any significant radiological release to the environment. It includes the capability of long-term cooling and excludes the occurrence of re-criticalities. It is achieved through maintaining the integrity of the main safety barriers such as main vessel by reducing the possibility and amplitude of power excursions. To comply with this purpose, prevention and mitigation strategies are developed.

III. PREVENTION ANF MITIGATION BY DESIGN

In the frame of the French SFR accident prevention, a specific core concept has been developed. The concept applies an upper sodium plena and an axially heterogeneous core that allows a less energetic primary phase due to the favorable sodium void worth characteristics of the core.

Concerning accident mitigation, an innovative strategy is being investigated. It consists of mitigation transfer tubes and a core catcher. The transfer tubes are envisaged to evacuate molten materials from the core center region where there is a potential for the formation of uncool-able reactive molten pools, Figure 2. By limiting the amount of fissile material in the core center region, the transfer tube is dedicated to decrease the probability and amplitude of power excursions. The role of the core catcher is to collect the arriving melt from the transfer tube, facilitate its spreading and provide long-term cooling [2].

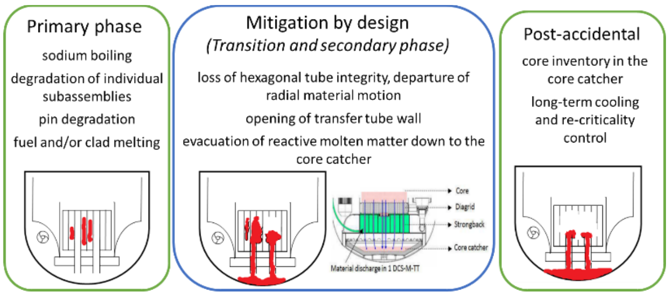

Figure 2. The concept of the mitigation transfer tubes.

A. *Physical phenomenon inside the transfer tube*

The transfer tube has the shape of a regular fuel assembly filled with liquid sodium during normal operation. It runs through the whole core, crosses the diagrid, penetrates the strong back and opens to the lower plenum where the core catcher is located.

During a severe accident evolution, the tube starts to perform its function when the neighboring hexagonal walls of surrounding subassemblies lose integrity and allow the containing high temperature molten mixture to touch and eventually melt the surface of the transfer channel. The tube opens to accommodate and guide the entering mixture downwards to the core catcher. The driving forces for the downward vertical material relocation are gravity and pressure. The downward pressure arises from energetic fuel-coolant interactions in which rapid thermal-to-mechanical energy conversion occurs when very high temperature molten fuel or cladding encounters relatively cold liquid sodium. The flow inside the transfer channel is a complex mixture of sodium vapor, solid bodies such as fragmented fuel pellets (3-10 mm) and smaller fuel and steel particles produced from fuel-coolant interaction (typically 0.05 mm), and liquid components such as melted steel and fuel as well as liquid sodium. Complementary to the complexity of the three-phase flow, there can be freezing or melting and vaporization or condensation of these components along the discharge path. Additional phenomenon related to the interaction between the mixture and the tube wall such as crust formation must be taken into account. Some phenomena, mainly the upward product of FCI, loss of flow mobility due to cooling, freezing, crust formation and the combined effect of them, have the potential to retain the relocation process. Thus, only a part of the entering molten mass may arrive to the lower exit of the tube and being discharged to the core catcher.

The success of the mitigation depends on whether sufficient fraction of fissile material can be evacuated from the core center region. Therefore, it is necessary to estimate the discharge efficiency inside an individual tube. After, the number and position of these tubes can be assessed as well.

IV. NUMERICAL SIMULATION

To evaluate the discharge performance with a high confidence, validated experiments or robust calculation tools can be used. The theoretical demonstration is currently based on best-estimate calculations with the mechanistic reference computer code SIMMER (Sn Implicit Multiphase Multicomponent Eulerian Recriticality). It is an advanced safety analysis computer code dedicated to investigate postulated core disruptive accidents in liquid-metal cooled fast reactors (LMFRs). The fluid-dynamics solution is based on a four-step time factorization time splitting algorithm that solves intra-cell transfers and convection terms in separate calculation steps [3]. SIMMER provides a numerical simulation tool based on mechanistic and physical models for complex multiphase, multicomponent flow problems that are fundamental to core destructive accidents.

Recent simulations of severe accident scenarios with the mitigation strategy have demonstrated, that the discharge of molten and degraded core inventory via a certain number of transfer tubes can be efficient. However, uncertainties of SIMMER approach are identified on the relocation process related to a possible total blockage formation inside the transfer tube due to particle-size debris accumulation. It is believed to originate from the solid particle treatment in the code.

As the material discharge efficiency, and therefore the prediction on reactivity removal, is related to the description on solid particle movement, there is a strong motivation to evaluate the accuracy of particle rich flow modelling in SIMMER. To identify modelling shortcomings, a review of particulate flow description in SIMMER is being performed.

A. *State-of-the-art SIMMER approach*

In SIMMER code, the treatment of particle motion is a user choice. However, the code validation is carried out with specific recommendations. The state-of-the-art description assumes that particles are suspended in their corresponding liquid phase and they always travel at the same speed as their carrying liquid form. In this approach, the viscosity of a liquid-solid suspension is increased compared to the viscosity the liquid alone, referred as apparent viscosity.

However, the validity of the apparent viscosity model becomes ambiguous when solid particle fraction approaches the maximum packing limit. Maximum packing is the highest density so the largest solid fraction attainable by spherical particles in a given volume. Around this limit, the mixture viscosity is exponentially increased due to the mechanical interlocking of particles, and thus further movement is prevented. In the calculation, flow blockage is observed. Physically, with increasing solid fraction, particle level interactions start to govern the flow characteristics and the repulsive forces may prevent blockage formations. To account for these effects, the assumption on suspended particles may not be the most accurate representation of the physical phenomena.

B. Detailed SIMMER approach for partciles

SIMMER, being a multi-velocity-field code, is capable to treat solid fragmented materials as an individual phase by assigning a separate set of balance equations to them. This approach is currently investigated in the French frame of research. In SIMMER, the simplified Navier-Stokes balance equations are uniform for each type of components independent of the phase considered. Our first interest is to focus on the purely dynamic description of particle rich flows, described by the momentum balance equation (1).

$$\frac{\partial(\rho v_q)}{\partial t} + \nabla(\rho v_q v_q) + \alpha_q \nabla p - \rho_q g + K_{qs} v_q - \sum_{q'} K_{qq'}(v_{q'} - v_q) - VM_q = -\sum_{q'} \Gamma_{qq'}\left[H(\Gamma_{qq'})v_q + H(-\Gamma_{qq'})v_{q'}\right] \tag{1}$$

It states that the change of momentum, first term, with time is equal to the net rate of momentum transferred by convection, second term; the acceleration due pressure gradient p, gravity force g, drag force of structures K_{qs} and other components $K_{qq'}$; virtual mass force VM_q and the momentum source of newly created mass $\Gamma_{qq'}$. The subscript q stands for velocity fields. See description and units in Appendix.

To evaluate in what contexts this uniform momentum equation is appropriate to account for the specific physics of particulate flows, it is necessary to examine the assumptions behind each term. In this context, the verification of the separate particle momentum approach in SIMMER code is launched. In the verification procedure, reactor scenarios associated to the material discharge via the transfer tubes are analyzed.

V. VERIFICATION OF VERTICAL FLOW OF PARTICLES

In reactor severe accident application during the discharge process from the core center zone, situations can develop in which fragmented core materials fall solely by gravitational effects in quasi-stationary surrounding fluid. The description of the purely vertical debris bed sinking in liquid or vapor sodium environment can be related to the concept of terminal velocity.

The terminal velocity is a fundamental concept in sediment transport processes. It states that any object moving through a viscous fluid eventually reaches an asymptotic velocity after travelling a critical distance. It is attained when the upward product of buoyancy and drag forces exerted by the surrounding fluid counterbalances the downward gravity force such that the net force on the body becomes zero. The balance of forces brings about zero acceleration leading to a constant rate of motion. The free fall velocity is the highest achievable speed of for the gravity driven motion of suspended particles.

Considering the momentum equation in SIMMER, this scenario is aimed to verify the drag force computed between solid particles and surrounding vapor. The drag force is computed assuming the similarity hypothesis, proposed by [4], for the drag coefficient on spherical bodies. The correlation implemented in the code is written as equation (2) with C_D being the drag coefficient and Re the Reynolds number.

$$C_D = 2.4Re^{-0.25} \tag{2}$$

To verify the accuracy of SIMMER computed terminal fall velocities, a simple tube filled with vapor and low fraction of solid particles that falling solely by gravity is simulated. Particle diameters are representative to reactor scenarios ranging from 0.05 mm to 3.5 mm, Figure 3.

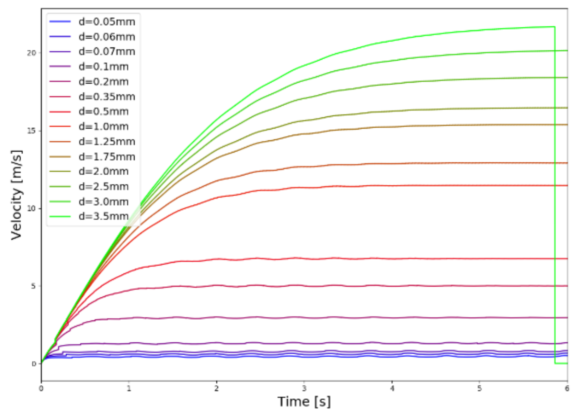

Figure 3. SIMMER predicted termial velocities as a fucntion of particle size.

The curves for each grain size are in agreement with the concept of the terminal velocity. After a certain distance travelled, a constant speed is reached. The balance between the drag and gravity force as a function of grain diameter is such that higher and more delayed asymptotic velocity is attained for larger grains. Theoretically, there are several empirical and analytical formula proposed to calculate solid's terminal falling velocity with particle diameter. Physics-based analytical equations for spherical particles include the Stokes and Impact laws. Empirical correlations include Rubey and later modified by Watson, Gibbs *et al.*, Cui *et at.* and Ferguson ad Church [5]. These expressions with their range of applicability are compared to SIMMER results (taking the asymptotic velocity for each size) of athermal solid particles of changing size falling in a vertical channel filled with vapor.

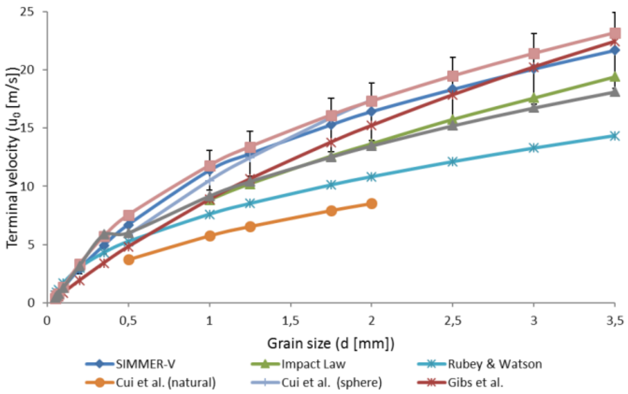

Figure 4. Comparison of terminal fall velocities.

From the comparison, Figure 4, it is clear that even empirical equations yield a broad range of terminal velocities. The observed large uncertainties arise from the limited opportunities for experimental data synthesis due to the lack of comparison methods. On a global scale, SIMMER results closely follow the curves of Ferguson & Church's model. Considering 6% r.m.s. error of SIMMER results, the equation of Ferguson & Church verifies the result for each grain size.

VI. CONCLUSIONS AND ORIENTATIONS FOR FUTURE R&D

In this paper, due to the importance of solid phase, mainly fuel, relocation during the severe accident mitigation scenario of SFRs, the impact of solid fragment treatment is studied through the SIMMER approach. As it was introduced previously, the particle motion within SIMMER code may be treated by separate balance equations. Although, the uniformity of these equations raises the question whether the specific physics of particle systems is well represented by these simplified Navier-Stokes equations and their closure terms. To evaluate SIMMER solid particle treatment, verification and validation activities have been launched. The first results are presented in this paper.

On the other hand, future R&D work is proposed to improve SIMMER modelling of particles by investigating granular flow models. Identifying the most prominent features of granular mater, additional terms can be introduced into existing SIMMER equations. Granular theories describe the behavior of large population of discrete macroscopic bodies, which can be associated to solid degraded core materials encountered during the mitigation scenario of SFRs. Granular matter under certain conditions displays similar behavior to solid, liquid and gas, although, the global behavior is different from any other forms of matter. There are different theories developed to describe each regime including the effects of particular granular features. The two essential aspects contributing to the unique properties are the dissipative nature of interactions between grains due to the static friction and inelastic collisions, and that the ordinary temperature plays no role in motion [6]. With the current state of overview, the liquid regime is the most suitable to describe the fragmented material discharge process. It is partially due to the continuum approach that are similar to SIMMER architecture, and thus would facilitate the implementation of the new model. Another reason is the correspondence between the characteristics of granular liquid and the relatively dense, continuously flowing nature of fragmented core materials inside the tube. Future R&D work is oriented towards the theoretical description of this regime.

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APPENDIX

Symbol	Meaning	Units
α	Void fraction	-
g	Gravity	m/s^2
H	Heaviside function	-
ρ	Density	g/cm^3
p	Pressure	Pa
v	Velocity	m/s

Innovative technologies and safety issue on the example of the world's first Floating Nuclear Power Plant (FNPP)

Dabizha Tatiana Grigorievna

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I. INTRODUCTION

In the process of development of civilian nuclear technologies and the appearance of a large number of nuclear reactors on military vessels, submarines and icebreakers, the benefits of mobile energy sources that can be used in remote and undeveloped areas have become apparent.

The development of projects of low-power reactor plants can effectively solve the problems of providing energy supply to distant areas with decentralized electricity supply and expensive fuel.

In Russia, Floating Power Unit (FPU), primarily is highly sought in the Far North and Far East which do not have a unified energy system and do need for the reliable and economically acceptable energy sources. Floating Power Unit (FPU) can be used also as a part of a desalination complex. Interest in such desalination complexes is manifested in many countries in Africa, Asia, Europe, experiencing an acute shortage of fresh water.

The Floating Power Unit (FPU) Akademik Lomonosov - is a unique and first-ever project of a low-power mobile transport power unit designed to operate as a part of the Floating Nuclear Power Plant (FNPP).

Despite the fact that civilian floating reactors were used in the United States of America to provide energy for the Panama Canal (1966-1976) and the American research base in Antarctica (1962-1972), today Russia remains the only country in the world that has mastered the *full production cycle* of floating nuclear power plants, and the Floating Power Unit (FPU) Akademik Lomonosov is officially the northernmost nuclear power plant in the world.

Floating Power Unit (FPU) Akademik Lomonosov is a *mass production project*, unlike the American Sturgis station, which in fact was not a special vessel, but was a converted bulk carrier Charles H. Cugle of Liberty type. The electrical power of Sturgis, which operated on nuclear fuel, was 10 megawatt against 70 megawatt of the FPU Akademik Lomonosov. Unlike the Russian FPU Academician Lomonosov, the American Sturgis station was used for military purposes and operated in the Panama Canal area. In 1976, the Sturgis was returned to the United States and was no longer in use.

All this conditions give the right to recognize the Floating Power Unit (FPU) Akademik Lomonosov as the first *specially developed civil design project of batch production*.

II. PRIMARY PURPOSES

The project itself is intended for reliable year-round heat and power supply for remote regions as Arctic and the Far East. The primary purposes of Floating Nuclear Power Plant is to provide energy to remote industrial enterprises, port cities, as well as gas and oil platforms located on the high seas.

In September 2019, Floating Power Unit (FPU) Akademik Lomonosov arrived from Murmansk to the northernmost Russian port of Pevek, Chukotka.

The launch of new Floating Nuclear Power Plant is aimed at solving of two key tasks:

1) replacement of the retired capacities of the Bilibinskaya NPP, operating since 1974, and the Chaunskaya co-generation Power Plant (CPP), which is more than 70 years old. The stopping times for the 1st unit of the Bilibinskaya NPP in 2019 are synchronized with the commissioning of the FNPP in the Pevek in Chukotka;
2) providing energy for the main mining companies located in Chukotka - a large ore-metal cluster, including gold mining companies and development projects of the Baimskaya ore zone.

Floating Nuclear Power Plant (FNPP) has electric power more than 70 megawatt. The electric capacity of each reactor is 35 megawatt. There are 2 Icebreaker type KLT-40S reactors installed on the Floating Power Unit (FPU) Akademik Lomonosov, the prototype of which are the reactors of the operating nuclear-powered icebreakers as «Taymyr» and «Vaygach» and the lighter carrier «Sevmorput».

Safety was the main priority during the construction of the FNPP, therefore, the power capacity of FPU was set up in stages, with the necessary tests of the plant equipment for its further safe operation.

Capacity of the Floating Power Unit (FPU) Akademik Lomonosov should compensate the losses from the decommissioning of the Bilibino NPP, built about half a century ago. The energy capacity of the Floating Power Unit (FPU) Akademik Lomonosov is 70 megawatt, twice more than the Bilibino NPP has (electric capacity of the Bilibino NPP is 48 megawatt (4 power units - 12 megawatt each).

Rosatom State Atomic Energy Corporation is already working on the second generation of floating nuclear power plants. It plans to optimize the floating power unit, making it smaller and more powerful. It is assumed that it will be equipped with two RITM-200M reactors with a total capacity of 100 megawatt.

The Floating Nuclear Power Plant (FNPP) Akademik Lomonosov as mentioned has the goal to displace coal-fired power plant and nuclear power plant, that will decrease the amount of carbon dioxide emissions into the atmosphere more than half a million tons.

When coal is burned, even at modern Thermal Power Plants (TPPs) operating on coal with an ash content of no more than 10% and equipped with a filtering system that allows 97.5% of ash to be retained, the emissions anyway almost completely enter into the external environment. As a result, the specific activity of the TPP`s emissions is 5-10 times higher than activity of the NPP`s emissions [1]. The fossil fuels burned at the TPP always contain radionuclides of natural origin. Operation of the Thermal Power Plant leads to higher emissions of radiation doses to humans rather than operation of the Nuclear Power Plant - in total almost 3 times more.

In addition, the Thermal Power Plants emit into the atmosphere much more carbon dioxide and chemical pollutants.

During normal operation of a Nuclear Power Plant, the amount of radioactive substances entering into the environment through gas aerosol emissions and liquid discharges is small.

The dose of external and internal exposure of the human body along the perimeter of border of the sanitary protection zone around the Nuclear Power Plant and beyond is much lower than the established standards.

Nuclear Power Plants are considered to be one of the cleanest ways to produce energy. The environmental load is further reduced by use of the Floating Nuclear Power Plants by minimizing construction work on the construction site of the earth territory and because of the possibility to dispose the Power Unit after its operation.

The nature and amount of gas aerosol radioactive emissions depend on type of the reactor and on system for handling this waste: inert radioactive gases and radionuclides.

Thus, the operation of the type KLT-40S reactors eliminates toxic emissions and air pollution. The radiation impact on the environment is limited to fractions of a percent of the level of natural background.

III. Technical Parametrs

The Floating Nuclear Power Plant (FNPP) Akademik Lomonosov includes:

1. floating power unit with two reactor units KLT-40S and two steam turbine units TK-35 / 38-3.4s;

2. hydraulic structures that ensure the installation and unfastening of the power unit and the transmission of generated electric and thermal energy to the shore;

3. onshore facilities designed to transfer the generated electric and thermal energy to external networks for distribution to the consumers.

Reactor and steam turbine units are located in the casing of the Floating Power Unit (FPU). Storage facilities for fresh fuel assemblies, spent fuel assemblies, solid radioactive waste and liquid radioactive waste, electric power system, automatic control system “Laguna”, general ship systems and equipment, as well as residential and office premises are also located in the Floating Power Unit (FPU) [2].

IV. Operating Safety

The Floating Nuclear Power Plant (FNPP) Akademik Lomonosov is the first of its kind, but nuclear reactors have been operating at sea since the first nuclear submarine was launched in 1955. Therefore, the Finnish Radiation and Nuclear Safety Authority (STUK) indicated in advance that transportation of the Floating Power Unit (FPU) Akademik Lomonosov along the Gulf of Finland "will not pose any problem" [3].

The FPU Akademik Lomonosov is designed with a large margin of safety, which makes nuclear reactors not vulnerable to tsunamis and other natural disasters. In addition, nuclear processes at a floating power unit meet all the requirements of the International Atomic Energy Agency (IAEA) and do not pose a threat to the environment [4].

Since the FPU Akademik Lomonosov is a floating, its casing and all its characteristics, including seaworthy, are considered to be the most important elements affecting the facility's safety, including nuclear and radiation.

The FPU Akademik Lomonosov is a flat-deck flat-bottomed non-self-propelled vessel of the stay in place type (i.e., its main operating mode is long-term berthing at the berth).

When two adjacent compartments are flooded, the floating structure will remain afloat without losing stability and buoyancy.

Operation of the installation can be carried out by one operator from the central control room without the constant presence of other maintenance personnel in the premises of the power equipment.

Each of the reactor units (RU) is enclosed in a steel sealed enclosure, designed as a solid-weight construction of the floating power unit housing and designed for the maximum pressure that can arise in it in case of emergency.

It is stated that the enrichment of uranium used in the cassette core does not exceed 20%, and this allows Rosatom to fulfill the IAEA conditions to limit the spread of highly enriched nuclear material and improve the investment attractiveness of the project (nuclear material with enrichment up to 20% is considered medium enriched).

Storage of liquid and solid radioactive waste should be carried out without the involvement of special nuclear-powered service vessels and floating technological reloading bases during 12-year overhaul period.

At the repair and maintenance plant, the nuclear fuel spent in one operational overhaul period (10-12 years) should be reloaded from the FPU storage facilities into transport containers and sent for reprocessing.

The casing of the FPU has ice reinforcements and special means for towing in the ice with an atomic icebreaker of the "Russia" type, as well as means for unfastening at the location.

The main casing and power structures of the superstructure are made of steel, which has a high resistance to brittle fracture at low temperatures. The underwater housing is protected against corrosion by electrochemical protection and ice-resistant paintwork [5].

Every 12 years, the FPU is towed to the factory for factory and dock repairs. During factory repairs, the fuel in the reactors is also reloaded and the radioactive waste discharged. One year is allotted for these works, after which the FPU resumes its work. In total, the estimated operating life of the FPU is about 40 years - three work cycles of 12 years with one-year interruptions for factory repairs.

According to the passport, the FPU should remain afloat in case of flooding of two adjacent compartments [5].

According to the project, safety of the FPU is ensured under a wind load at a wind speed of up to 25 m / s, a 7-8-magnitude earthquake, the fall of a light aircraft of the YAK-40 type, a lightning strike, an external source explosion on the shore or on docked FNPP, etc [4]. It is said in the project that the design of the FPU guarantees operability of the equipment, mechanisms and systems with an impact load of at least 3g, acting in any direction, under the conditions of tilting and rolling.

The safety level of these facilities is estimated by the developer (Rosatom) as high [5].

V. Conclusion

British newspaper The Independent announced: "Many supporters of renewable energy and energy-saving technologies say that their projects will provide future technological development. And only a few of them understand that significant technological progress is still possible in nuclear power" [6].

In addition to generation of the electric and thermal energy, Floating Energy Block has a water desalination complex that produces not only electricity, but also high-quality drinking water from seawater using either reverse osmosis (RO) technology or Multi-effect distillation (MED).

Among other things, the disposal of spent nuclear fuel by the stationary Nuclear Power Plant requires so much money funds that often this is equivalent to building a new one station. Therefore, in many remote regions, as northern latitudes for example, it is advisable to use Floating Power Units. As soon as they have been worked out, they will be towed to the specialized ports, where there are all necessary capacities for disposal. All of the above undoubtedly confirms the advantage in use of the FPU in the near future.

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Prospects of carbon-free electricity delivering civil submarines for disaster relief

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I. Introduction

Nations worldwide are annually threatened by life-threatening disasters such as hurricanes, flooding, droughts, wildfires and earthquakes [1]. Following these events, there is a high risk of power outage, inducing a paralysis of economic and social statures in affected regions. The current relief response is to deploy fossil fuel generators to these regions [2]. However, two main problems arise when taking this route: (1) unsustainability due to large carbon load associated with operation, (2) long (up to months) and often delayed delivery times. Access to electricity is important for critical infrastructure and many essential services such as water, gas, communications etc. [1]. For instance, more than 1 million people in Puerto Rico lost electricity in the largest black-out in the United States (U.S.) that spanned 11 months after hurricane Maria in 2017[3]. The lack of reliable electricity during the recovery time claimed the lives of approximately 2,900 people and led to an estimated $3 billion dollars in economic losses [3]. Worldwide, absolute losses incurred by natural and man-made disasters are even higher. Between 1994 and 2015, the economic loss by the Americas, Asia, Europe, Oceania and Africa was estimated at $870 billion, $709 billion, $262 billion, $40 billion, and $10 billion, respectively [4]. Scientific evidence suggests that climate change will get worse, thereby, continuing the trends in increasing severity and number of disasters worldwide.

II. Energy Sources

Reluctance to expand to low-carbon electricity generation is the most principle reason why the world falls short on international climate targets [5]. In 2017, nuclear energy provided 56% of America's carbon-free energy, followed by hydro at 21%, then wind, solar and geothermal at 18%, 4% and 1%, respectively [6]. An evaluation of capacity factors of different energy sources revealed that nuclear power plants operate at more than 92%, twice as much as coal (54%) and natural gas (55%) and 2-3x more than wind (37%) and solar (27%) [6]. Although nuclear energy has been the largest source of low-carbon electricity for >30 years, its future is uncertain as ageing plants are being shut down [5]. The construction of new land-based nuclear reactor projects is often crippled by cost overruns and delays [5]. It is of interest to explore innovative ideas to reduce construction costs of nuclear technology while expanding global use of carbon-free energy applications.

Innovations within the nuclear community can help to provide an approach for catastrophic black-out planning, response and recovery after natural disasters. With current advancement in reactor technologies, we need to utilize the offshore modular reactor technologies to provide much needed low-carbon solution for disaster relief [7]. Nuclear submarine technologies are well known within the Naval military around the globe with more than 40 years of operation experience [7]. Interestingly, nuclear energy generation began with ocean-based applications; however, in the early 2000s the ocean-based application of nuclear energy was limited to military implementation [7], [8]. These submarines rely on nuclear power for both propulsion and life support. Primarily used for naval military missions, nuclear submarines are not utilized for non-military efforts. By adopting and restructuring related technologies for the development of civil nuclear submarines for disaster relief, annual socioeconomic losses can be mitigated. This contribution assesses the prospects of the development and promotion of civil submarines in the peaceful uses of nuclear energy.

III. Offshore Nuclear Reactor Technology

Since the Fukushima Daiichi nuclear accident, efforts are being made by the nuclear community to mitigate consequences of natural disaster-induced accidents [7]. One very popular effort is to advance the nuclear material technologies by developing accident tolerant fuels and structural materials. Another effort is the investigation of offshore nuclear reactor (OFNR) technologies [7], [8]. The development of the floating reactor project in Russia began in the early 2000s. By 2010 the

OFNR Akademik Lomonosov driven by KLT-40s was launched. In the U.S., a new OFNR concept is being investigated on a spar-type floating platform [7]. The China General Nuclear Group (CGN) plans to complete its construction of a similar floating-type reactor with ACPR50S design by 2020. Meanwhile in France, Flexblue, a submerged-type OFNR was proposed by Naval Group (previously known as DCNS) in 2011 [7]. Flexblue was an immobile nuclear reactor concept designed to produced 50-250MW of nuclear energy while residing 60-100m underwater and up to 15km away from the shore. A prototypic Flexblue reactor was set to be built in 2013 for launch and deployment in 2016 [7]. However, the project was placed on hold and not realized to fruition. Compared to a floating reactor, a submerged nuclear reactor concept is more resistant to natural disasters such as flooding, earthquakes and tsunamis, which enforces its safety features associated[9].

(2019 Nuclear Innovation Bootcamp, French Alternative Energies and Atomic Energy Commission (CEA) and the Department of Materials Science Engineering, Nuclear Engineering Program, University of Florida)

IV. MOBILE CIVIL NUCLEAR SUBMARINE

The deployment of mobile civil submarines offers a carbon-free solution to disaster related problems worldwide. The mobilization of a submerged-type OFNR reactor similar to the Flexblue design would prove extremely useful for relief applications should this concept be adapted. A self-powered mobile nuclear submarine offers the opportunity to support infrastructures by traveling to countries experiencing blackouts. Given that some countries have not yet been introduced to nuclear energy technology, this presents an opportunity to globalize peaceful uses of nuclear. For example, the power generation sector in the Caribbean receives >70% of its energy production from imported fossil fuels [10]. Nuclear energy is not used in these regions, even though Jamaica is currently the only Caribbean island that operates a SlowPoke nuclear research reactor at the University of the West Indies [11]. The development of mobile civil nuclear submarines that can be shared among neighboring countries would cut down construction costs of brick and mortar power plants on different islands.

SubR is a proposed carbon-free electricity delivering civil submarine that utilizes the functionality of small modular reactors (SMRs) primarily for natural disaster relief but can also be used on demand as a clean and reliable energy source. The main reactor characteristics is similar to those for Flexblue [7], [9]. The submarine will contain a PWR unit and is projected to be 70m long and 12m wide producing 50MW of electricity for energy supply and self-propulsion. The proposed dimensions were selected in accordance with existing military submarine technology (typically 138m by 12.5)[12]. However, the proposed dimensions of SubR is reduced compared to the military design because of the space occupied by submarine-launched ballistic missiles are dispensable to the safe operation of SubR. Although to a lesser extent, the civilian design of SubR is similar to military designs such that priority is placed on components such as mobility and fuel lifetime. Dissimilar to military submarines, the strategic focus of SubR will be on fuel economy due to its different shape, lower operation cost. A typical commercial reactor produces 1GW of energy[6] enough for ~1 million people provided that one person uses 1kW on average each year. Following a disaster, the one SubR unit can produce 50MW of energy for 200,000 people.

The core will use low enriched uranium UO_2 (17×17 rods in square assembly) and Zr alloy cladding. Figure 1 depicts a schematic of the process by which electricity is generated and transported to homes/facilities (such as hospitals) via cables connected the submarine once it surfaces. The proposed technical parameters of the SubR design is summarized in Table 1. The main components of the SMR assembly are the pressurizer, steam generator, control rod and the core. The assembly is divided into a 3-loop coolant system for heat transfer. The primary loop transfers heat generated from the fuel to the steam generator. The steam generator then converts the water in a secondary loop into steam to drive the turbine for electricity generation. The third loop functions to help dissipate excess heat through the hull of the submarines by natural convection. Electricity generated from the turbine connects to generators that can either supply electricity to land or propel the submarine. The boat is equipped with a propeller is used to transport the cable(s) to and from the submarine. The boat exclusively serves as the cable installment and connection component. The service facility serves as a docking station for maintenance needs and upkeep of the submarine. Coastal regions interested in SubR's mobile civil reactor technology will be expected to accept the maintenance facility service to

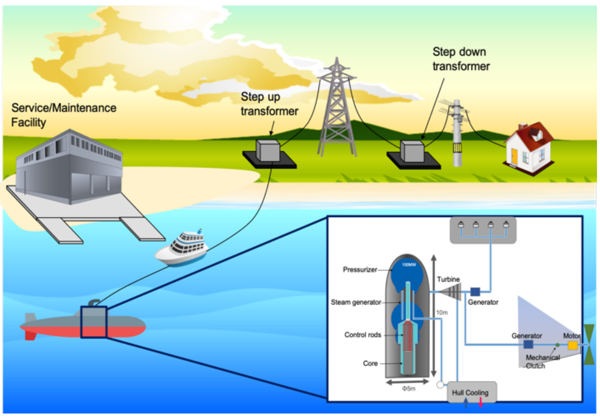

Figure 1. SubR's design concept means by which electricity generated and transported to homes/facilities (such as hospitals) via cable connectors.

ensure safe and efficient operation of the submarine. In the event that the boat is nonoperational, the support option is to directly connect the cables from the submarine to the maintenance facility at the docking station. In case of emergency, such as immediate need for a reactor shutdown, a lead acid battery can be used to convert DC power to propulsion until the submarine reaches a depth at which a snorkel could be deployed to run the diesel.

Table 1. Proposed technical parameters for SubR

Parameter	Value
Electrical capacity	50MW(e)
Thermal capacity	175MW (th)
Fuel enrichment	<5%
Core inlet/exit temperatures	288/318°C
Steam Pressure	15.5MPa
Coolant/moderator	Light water
Fuel cycle	30 months

There are several benefits to this type of technology as it: (1) aims to supply power various countries and/or islands that are annually threatened by natural disasters; (2) provides the opportunity for the globalization of nuclear energy as an alternate energy source for countries/regions that do not currently utilize this technology; (3) can be used to propel the advancements of desalination and hydrogen production technology efforts; (4) cuts down on nuclear reactor construction costs; (5) will utilize low-enriched uranium fuel to mitigate risks of nuclear proliferation; (6) is a self-powered nuclear submarine that has the ability to operate for up to 48 months without refueling. Furthermore, this project could also assist in response to industrial disasters. For example, one of the problems after the Fukushima accident is the pumping of radioactive water from the reactor to a filtering system housed inside a building the size of a small aircraft hangar. SubR offers a solution that uses nuclear power to remove and filter water contaminated with radioactive isotopes.

Some potential challenges of SubR are that: (1) localized heating in the ocean may affect marine environments and (2) the formation of joint government policies might be difficult to navigate. These challenges will not have a significant impact on feasibility of the proposed work. Firstly, the SubR is technologically feasible since nuclear submarine development is not a new topic. Secondly, impact on marine environments should be minimal, although more studies are needed to confirm this hypothesis. This disadvantage is negligible in view of the harm caused by disasters. Thirdly, International partnership for peaceful uses of nuclear will aid in solving policy issues. Extending the technology for peaceful uses of nuclear energy for disaster relief is a solution that can be implemented in the near future. To maximize the potential of the impact of a mobile civil nuclear submarine for disaster relief, government partnerships are recommended. Nations worldwide have annual disaster preparedness and relief budgets for national and international aid. Expanding these collaborations to aid via mobile civil submarines can prove effective in preventing loss of life and economic paralysis. In terms of safety regulations and public acceptance, the concept is new, similar to the immobile Flexblue concept. Licensing of a civil submarine could present challenges with different regulatory and licensing companies is different countries. Research and development on the feasibility, safety margin and cost of developing a prototypic SubR model is an ongoing effort.

CONCLUSION

In the search for viable carbon-free solution to minimize the catastrophic impact of disasters, a mobile civil nuclear submarine would provide the opportunity to: (1) save and or improve the lives of people affected by natural disasters; (2) help mitigate the impact of climate change; (3) reduce billions of dollars in economic losses to regions due to blackouts; and (4) globalize and promote cooperation in the peaceful uses of nuclear energy.

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Study on Mitigation Measures of MSLB under Load Following without Boron Adjustment Operation Strategy

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1. Introduction

Employing the Mode-C operation and control mode that performs load following without boron adjustment, the ACP600 3rd generation nuclear power unit is capable of automatically tracking boron load with the help of a rod control system, during which there is no need for frequent boron adjustment. Therefore, not only are wastewater treatment cost and operators' workload reduced, but the load following range can be extended from 80% to at least 95% in its lifecycle.

Nevertheless, it is more difficult to achieve power distribution flattening due to the operational mode of load following without boron adjustment. Consequently, worse consequences of the accidents may strike. Particularly, core power distortion during a main steam line break (MSLB) can be more significant in a context of load following without boron adjustment, even possibly threatens the safety of a reactor core. Hence, it is of great necessity to study measures that can be taken to alleviate MSLB accidents adopting load following without boron adjustment so as to relieve the consequences from MSLB accidents on the one hand, and satisfy the relevant design criteria as well as provide feasible suggestions on design optimization on the other hand.

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2. Overview of ACP600

Based on the No. 1 and No. 2 power units in Hainan Changjiang Nuclear Power Plant, ACP600 is designed with the experience from the Fukushima nuclear accident for reference, benchmarking 3rd generation nuclear power technical parameters and adopting the strategy of load following without boron adjustment. In addition, advanced fuel subassemblies independently developed in China are also utilized together with an emergency water supplementary plan, etc. Compare ACP600 and the engineered systems of No. 1 and No. 2 power units mentioned above, their main differences lie in the removal of both a high pressure safety injection system and a thick boron tank. As for major design parameters of ACP600, they as are listed in Table 1.

TABLE I. Main Design Parameters of ACP600

Parameters	Numerical values
Operational mode	**Load following without boron adjustment**
Rated thermal power of reactor, MWt	**1936**
Design life of plant, a	**60**
Design flow of thermal performance , m3/h	**46640**
Average temperature of reactor coolant , ℃	**310**
Operating pressure , MPa	**15.5**
Safety injection measure	**Medium pressure safety injection + Low pressure safety injection**
Boron concentration of refueling water tank, ppm	**2100**
Volume of safety injection box, m3	**47.7**
Operation pressure of safety injection box , MPa	**4.24**

3. Analysis on MSLB accidents and Mitigation measures

3.1. Analysis on MSLB accidents

After an MSLB accident takes place, SG heat exchange capability sharply rises as steam flow in a steam generator is rapidly discharged; consequently, both coolant temperature and pressure of the primary loop drop abruptly. Moreover, negative feedback of moderator temperature may cause the reactor core to return to its critical state , burn up fuel units and endanger the core. Considering this, a design criterion for MSLB core is designed as DNBR greater than 1.20 (FC).

In comparison with a full-power coolant system, energy stored in the hot showdown is much less, while its secondary side pressure and water capacity are much larger. For this reason, MSLB accidents take place when power of the hot shutdown is at its lowest level. In this study, analytic demonstration was carried out for MSLB specific to the original plan of ACP600. According to the analysis results, it is unlikely to prevent the reactor core from returning back to its critical state in an MSLB accident taking measures as medium pressure safety injection and low concentration boric acid solution. Moreover, the peak power of the reactor core that restores to its critical state reaches 32.46%, the minimum DNBR is 1.11, which fails to meet the design criteria and makes it likely to burn down fuel kits.

3.2. MSLB Mitigation measures

3.2.1. Boron Injection

(1) Incorporation of High Pressure Safety Injection

Based on the existing configurations of ACP600, a measure of injecting high concentration boron at high pressure is taken into consideration. As assumed, the injection of boron concentration is 7,000 ppm; and, the total injection flow was designed at 12m3/h, 24 m3/h, 48 m3/h and 96 m3/h for operating conditions 1-1, 1-2, 1-3 and 1-4 respectively. Then, corresponding analysis was made.

In Figures 1 & 2, time-varying curves of reactor core heat flux and pressure are presented in a transient process. The analytical results shows that peak power of the reactor core restores back to its critical state in an MSLB accident can be suppressed to a certain extent when boron injection flow rate is rather high. However, such a measure exerts a limited effect once the flow rate is low.

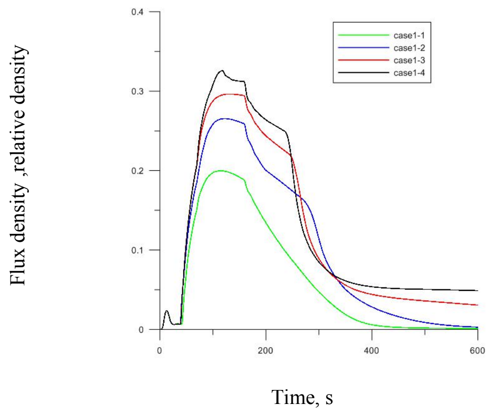

Figure 1. Reactor Core Flux Density.

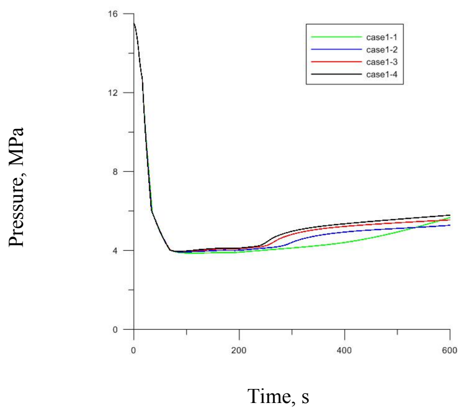

Figure 2. Reactor Core Pressure.

The study found that boron concentration in the reactor core before the flux density reaches its peak in the operating condition 1-1 slightly differs from that in a standard operating condition. By comparing safety injection flow rates of such two operating conditions, it was found that high pressure safety injection leads to high reactor core pressure that can limit the injection rate of the advanced accumulator. Additionally, its injection rate is dramatically greater than that of high or medium head safety injection,as shown in Figure 3. Therefore, it should be further explored whether accident alleviation can be realized by virtue of the accumulator only without high or medium pressure safety boron injection.

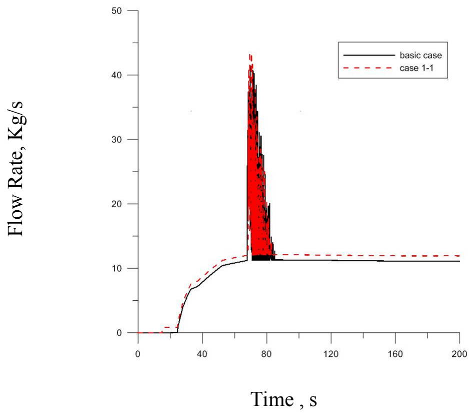

Figure 3. Safety Injection Flow Rate.

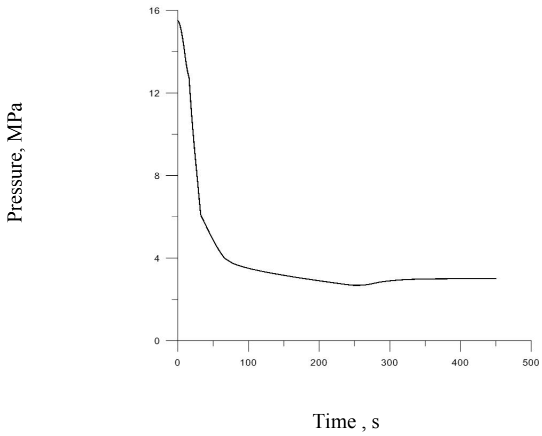

Figure 5. Reactor Core Pressure.

(2) Advanced Accumulator Only

On top of the standard framework, only the advanced accumulator is employed to alleviate MSLB accidents. Time-varying curves of core power and pressure in a transient process are presented in Figures 4 & 5 in such operating conditions. It has been proven by relevant analysis results that, regardless of high or medium head safety injection of boron, the peak flux density of the reactor core is 26.02% under the circumstance of an MSLB accident when only an advanced accumulator is adopted; in this case, the minimum DNBR is 1.29, therefore corresponding design criteria.

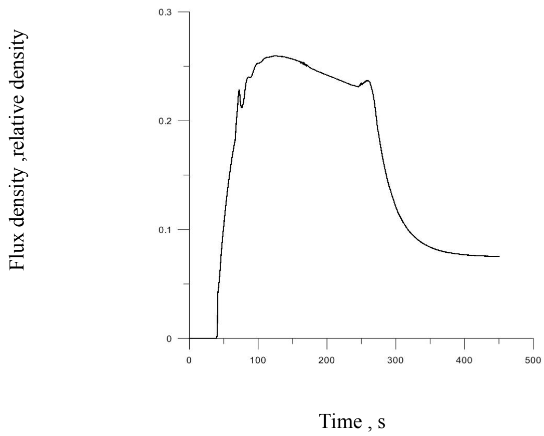

Figure 4. Reactor Core Flux Density.

The advanced accumulator can be put into use passively only when the primary side pressure is less than 4.24MPa, break characteristics calculations are conducted to analyze whether the advanced accumulator can be put into operation when accident take place under the circumstance that a steam pipe line break is rather small. Relevant analytical results indicate that the most restrictive operating condition is known as double-end shear fracture of steam lines (see operating condition 2-0) as far as MSLB accidents in an operating condition of hot shutdown are concerned.

The analysis of aboron injection into the reactor core to mitigate consequences of MSLB shows that: approach of high boron concentration by high pressure safety injection , the peak power of the reactor core that restores to its critical state during the MSLB accident can be suppressed to a certain extent when the injected boron flow rate is comparatively high. However, in case of a low flow rate, such a method exerts very limited effects. Under the circumstance that safety injection is eliminated, and only an advanced accumulator is utilized to inject boron, MSLB accident can be effectively retarded. In this event, the minimum DNBR is up to acceptance criteria. However, settings the signals of safety injection system should avoid the influence to other accidents, there is a great difficulty in configuring such signals.

3.2.2. Shut Down the Main Pump

Figure 6 is the reactor core heat flux in the MSLB accident when the main pump of the fault loop is shut down , it shows that main pump shutdown for the damaged loop plays a role in effectively mitigating MSLB accident consequences. Even if double-end shear fracture of the main steam line takes form, power of the reactor core that restores to its criticality is only 15% and the minimum DNBR is 1.29. This conforms to design requirements. Additionally, such a measure is also taken to cope with different breaks.

As for the relevant analysis results, please refer to Table 2,which shows the specific to MSLB accidents concerned with diverse break sizes, the damaged loop still can be effectively identified so that the main pump of this loop can be shut down timely. In this manner, MSLB accident consequences are validly mitigated.

TABLE II. BREAK CHARACTERISTICS ANALYSIS RESULTS -CASE 3

Case	Break area, ft2	Triggering time of "Low Steam Generator Pressure-3" pump shutdown signal, s	Peak power of main pump in service (in 5%)	Peak power after main pump shutdown for the damaged loop, %)
3-0	Double-end shear fracture; and flow area of a flow restrictor is 1.184	28	32.46	15
3-1	1	35.6	29.27	14.14
3-2	0.7	57.8	24	12.4
3-3	0.6	70.85	21.88	11.47
3-4	0.5	88.8	19.37	10.12

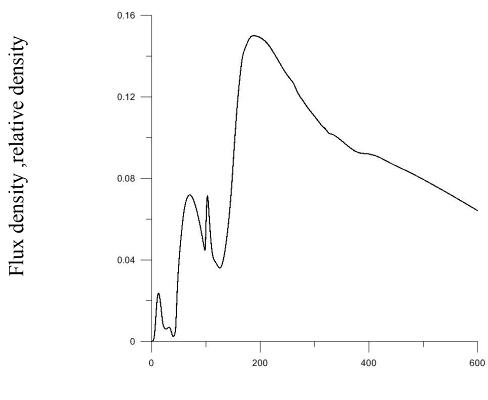

Figure 6. Reactor Core Flux Density.

This study shows that, main pump shutdown after an MSLB accident takes place has the potential to not only effectively reduce heat convection between the primary and the secondary sides for a steam generator of this loop, but also lower positive reactivity introduced into the reactor due to a temperature effect of the moderator, consequences of the MSLB accident are eventually mitigated.

4. SUMMARY AND SUGGESTIONS

According to to the investigation for MSLB accident mitigation measures under the load following without boron adjustment operation, the analysis and recommendations for suggestions are as follows:

1) MSLB accident mitigation measures under the load follow without boron adjustment operation strategy might burn down to the fuel elements and cause the damage to the reactor core.

2) Boron injection into the reactor core plays a limited role in relieving MSLB accident , and has a great influence on the original ACP600 design. Comparatively, shut down the main pump in the damaged loop is able to significantly mitigate consequences of the MSLB accident. Moreover, the latter scheme influences on the original ACP600 design is minor.

3) It is suggested that the "Low SG Pressure-3" signal be configured to trigger the main pump shutdown, so that consequences of the MSLB accident can be mitigated. However, a further study should be subsequently conducted to investigate the influence of such a signal on other accidents and systems on the one hand, and comprehensively evaluate feasibility and rationality of such a configuration on the other.

In this paper, measures for MSLB accident mitigation were preliminarily explored by taking an operating strategy of load following without boron adjustment into account. This study can provide references and suggestions for optimizing capacity to deal with ACP600 related accidents.

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TRACK 3: NEUTRONICS AND REACTOR PHYSICS

A STUDY OF TRANSVERSE BUCKLING EFFECT ON THE CHARACTERISTICS OF BURNUP WAVE IN A DIFFUSIVE MEDIA

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INTRODUCING THE FLOATNUC PROJECT: THE KLT – 40S FLOATING NUCLEAR POWER PLANT SIMULATION AT VARIOUS SEA STATE CONDITION

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PRELIMINARY NEUTRONIC DESIGN OF HIGH BURNUP 15 TIMES RECIRCULATION CYCLE PEBBLE BED REACTOR

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AN ARTIFICIAL NEURAL NETWORK SOLUTION TO THE POINT REACTOR KINETICS EQUATIONS WITH ONE PRECURSOR GROUP

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MACHINE LEARNING BASED METHODOLOGY FOR ASSESSMENT OF DOPPLER REACTIVITY OF SODIUM COOLED FAST REACTOR

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A Study of Transverse Buckling Effect on the Characteristics of Burnup Wave in a Diffusive Media

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I. INTRODUCTION

In nuclear reactors, diffusive media is considered as the mixtures of neutron absorbers with moderators. These diffusive media are applicable in control rods, reflectors and in fuel mixture of reactors. The reactivity worth of control rods decreases during the operation of the reactor due to burnable poisons. Conversion of absorbing material into non-absorbing one by neutron capture and the decrease in reactivity happens at the same time. If this effect of fuel and poisons on reactivity can be balanced by design, flattening of the reactivity to time curve can be obtained. If there is a strong neutronic coupling between the reactor core and reflector like in graphite-moderated reactors, burnable poisons can be used in reflector regions to obtain reactivity flattening. The fuels used in proposed CANDLE (Constant Axial shape of Neutron flux, nuclide densities and power shape During Life of Energy producing reactor) [1] reactor are mixed with burnable poisons for self-regulation and inherent safety. It helps the reactor to remain subcritical in initial configuration. Once poisons absorb neutron from fission wave, the region will become critical allowing the power producing fission reaction zone to pass through. So the study of diffusive media helps in the design of above reactor component.

Previous studies are there on the development and characterization of burnup wave on absorbing and diffusive medium [2],[3],[4]. The present work is extended to cylindrical diffusive medium to study the effect of transverse buckling led radial neutron leakage on the characteristics of burnup wave.

II. MATHEMATICAL MODELING

A two-dimensional cylindrical diffusive medium is studied. Diffusive medium is having Graphite as moderator and Boron-10 as burnable poison. Neutron flux distribution and burnup characteristics of the medium are calculated using the neutron diffusion equation and burnup equation. Diffusion equation for two-dimensional cylindrical system can be considered as,

$$D\frac{\partial^2 \emptyset}{\partial z^2} + D\frac{\partial^2 \emptyset}{\partial r^2} + D\frac{1}{r}\frac{\partial \emptyset}{\partial r} - \sigma N\emptyset = \frac{1}{v}\frac{\partial \emptyset}{\partial t} \qquad (1)$$

Where time derivative term of the neutron flux on the right-hand side of the equation is neglected as variation of flux is very slow compared to the neutron velocity. So the diffusion equation becomes,

$$D\frac{\partial^2 \emptyset}{\partial z^2} + D\frac{\partial^2 \emptyset}{\partial r^2} + D\frac{1}{r}\frac{\partial \emptyset}{\partial r} - \sigma N\emptyset = 0 \qquad (2)$$

The neutron leakage term of Eq.2, i.e., $\mathrm{D}\frac{\partial^2 \emptyset}{\partial \mathrm{r}^2} + \mathrm{D}\frac{1}{\mathrm{r}}\frac{\partial \emptyset}{\partial \mathrm{r}}$ can be replaced by employing transverse leakage approximation. Radial leakage is $-\mathrm{B_r}^2\emptyset$ under this approximation [5], where $\mathrm{B_r}^2 = \left(\frac{2.405}{\mathrm{R}}\right)^2$ is the geometrical buckling and R is the radius of the cylinder. The modified diffusion equation is,

$$D\,\frac{\partial^2 \emptyset(z,t)}{\partial z^2} - \left(DB_r^2 + \sigma N(z,t)\right)\emptyset(z,t) = 0 \qquad (3)$$

The Burnup equation for poison nuclide density is,

$$\frac{\partial N(z,t)}{\partial t} + \sigma N(z,t)\emptyset(z,t) = 0 \qquad (4)$$

Where,

D: Diffusion coefficient of the medium
v: Neutron velocity of the medium
σ: Microscopic absorption cross-section of burnable poison
∅: Space and time-dependent neutron flux
N: Space and time-dependent burnable poison nuclide density
z represents the axial direction

Reaction rate density is given by,

$$R_x = \sigma N(z,t)\emptyset(z,t) \qquad (5)$$

Initial and boundary conditions of this problem are considered as,

$$\mathrm{J(0,t) = J_0,\ N(z,0) = N_0,\ \emptyset(z_{end},0) = \emptyset_0}$$

Where, J is the neutron current density and $\mathrm{J_0}$ is the input current density applied on the left end of the cylindrical medium at z=0. $\mathrm{N_0}$ and $\emptyset_0$ are the initial poison nuclide density and neutron flux respectively. Flux is assumed to vanish at the boundary. Graphical representation of the problem is given in Fig. 1.

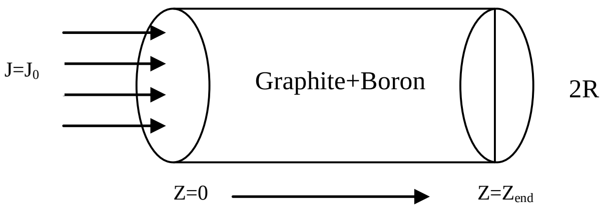

Figure 1. Two-dimensional boron+graphite cylindrical rod

Input data required for the problem are used from [2] and [4]:

1. Net input neutron current density (J_0) $=10^{14}$ cm^{-2}s^{-1}
2. Diffusion coefficient (D) = 1cm
3. Boron-10 density (N_0) = 1.04 x 10^{19} cm^{-3}
4. Boron-10 absorption cross-section (σ) = 3847 barns
5. Slab length = 100 cm
6. Initial flux of the medium $\frac{J_0 L}{D})$ = 5×10^{14} cm^{-2}s^{-1}

Diffusion length L = 5cm calculated using the equation,

$$L = \sqrt{\frac{D}{N_0 \sigma}} \qquad (6)$$

Finite difference method (Central Difference scheme) and fourth order Runge-Kutta method [5], [6] is used to solve the diffusion (3) and burnup eq. (4) respectively. MATLAB programming environment is used to carry out the simulation.

III. Results And Discussion

Transient and steady state parameters that are used to characterize the burnup wave are described below.

1) *Transient phase*

Transient Time (TT): Time required attaining the asymptotic (steady) state of the burnup wave. Generally value of the parameter is taken at the time when wave attain about 99 percent of its asymptotic (steady) state value.

Transient Length (TL): Distance covered by the burnup wave to attain asymptotic (steady) state value. Transient length is also defined by the value at which wave achieved 99 percent of its steady state.

2) *Steady State Phase*

Velocity of Wave Propagation: This is basically the burnup wave velocity.

Reaction Rate Zone Width:

- Full Width Half Maximum (FWHM): Width of reaction rate curve between the half values of the maximum reaction rate.
- Full Width 1% of the Maximum (FW1M): Width of reaction rate curve between 1% values of the maximum reaction rate.

Results for the time evolution of neutron flux, boron-10 nuclide density and reaction rate have been obtained for different values of radius (100 cm, 200 cm, and 1000 cm) of the cylinder.

Fig. 2 , 3 and 4 shows the spatial distribution for radius of the cylinder =100 cm. Each curve in these figures is plotted at an interval of 10 days and covers a period of 180 days.

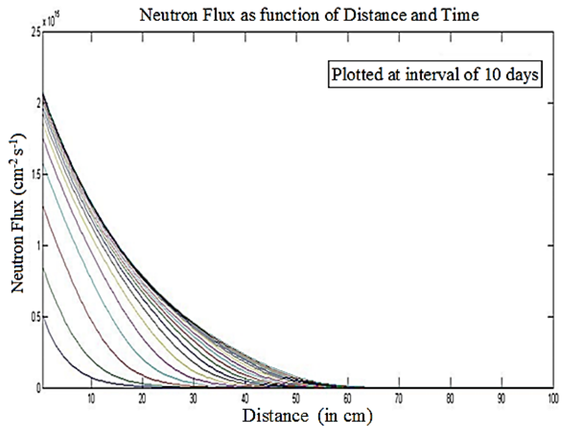

Figure 2. Neutron flux vs. space for radius = 100 cm

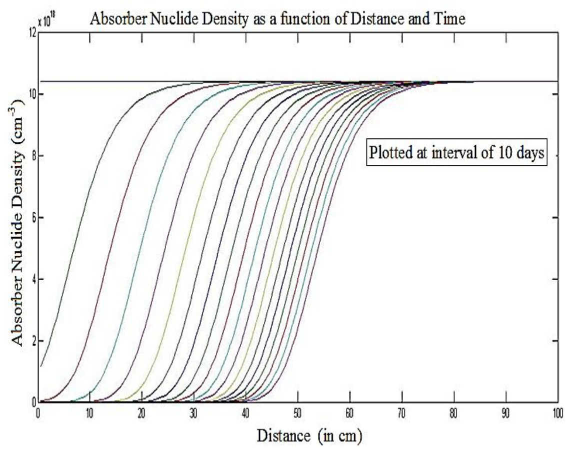

Figure 3. Absorber nuclide density vs. space for radius = 100 cm

Fig.5 shows the Maximum reaction rate with respect to time for radius 100 cm.

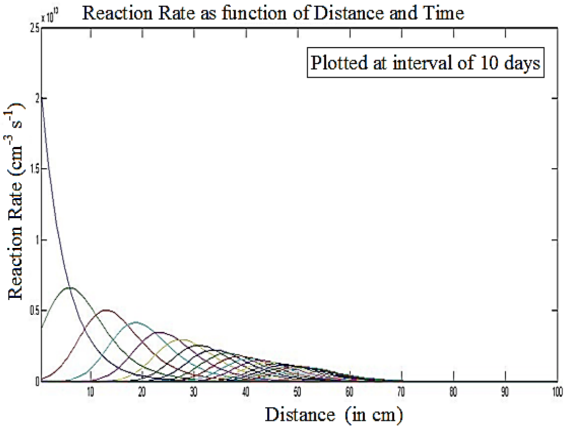

Figure 4. Reaction rate vs. space for radius R= 100 cm

It can observe that leakage is high and asymptotic state of the burnup wave is reached but due to neutron leakage, reaction rate decreases with time in space.

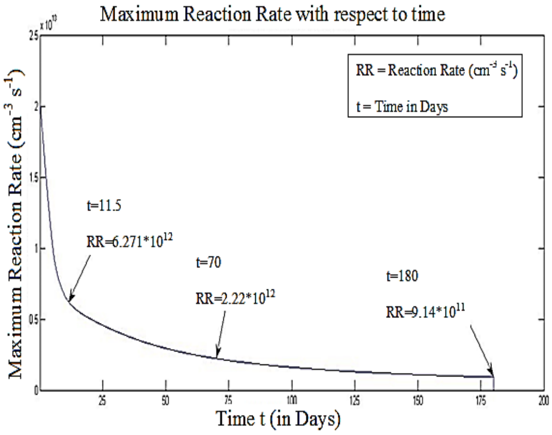

Figure 5. Maximum reaction rate for radius = 100 cm

Spatial profile of neutron flux, absorber density and reaction rate for radius = 200cm depicted in Fig. 6, 7 and 8 respectively.

Fig. 6 shows that with increase in the radius of the cylinder, neutron flux is high due to reduction in neutron leakage but it still does not reach the right end of the cylinder. Fig. 7 shows that the absorber nuclide density could not consumed the right end side of the medium. Fig. 8 shows that the reaction rate could not sustain up to the full length of the cylinder, means burnup wave could not propagate throughout the whole length. Fig. 9 indicates that the maximum reaction rate keeps on decreasing as a function of time but FWHM remains almost constant.

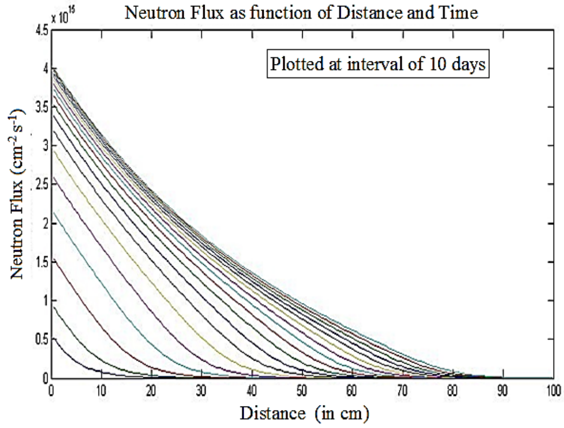

Figure 6. Neutron flux vs. space for radius = 200 cm

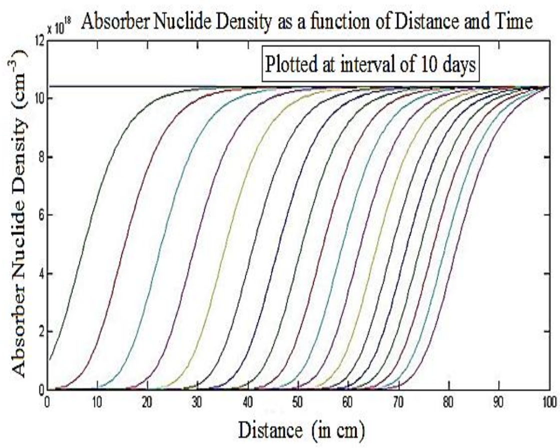

Figure 7. Absorber nuclide density vs. space for radius = 200 cm

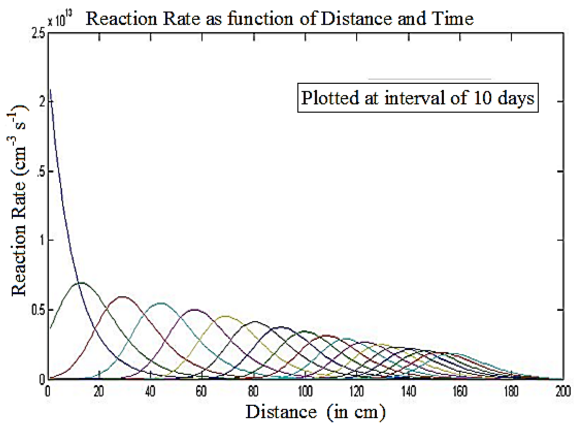

Figure 8. Reaction rate vs. space for radius = 200 cm

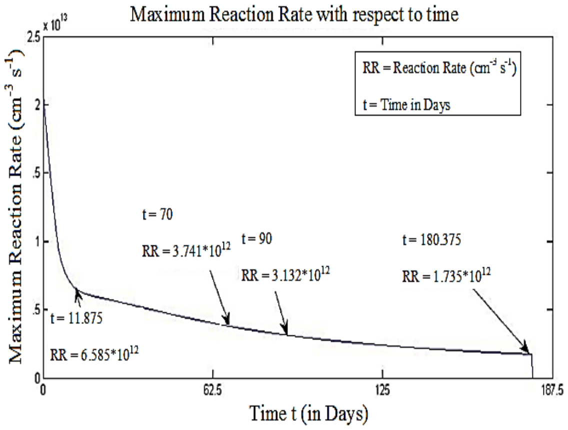

Figure 9. Maximum reaction rate for radius = 200 cm

Results for the case when radius is 1000 cm are plotted in the Fig. 10 respectively. This is not a practical case and this case is studied to check the extreme effect. This case can be considered as infinite medium. Only the spatial profile of reaction rate is depicted in Fig. 10. As we can see burnup wave propagate full length of the cylinder.

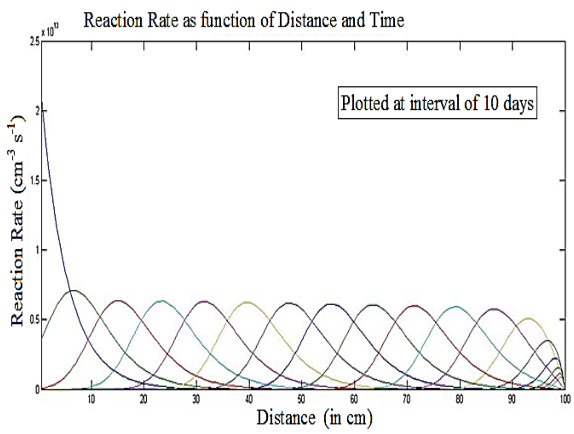

Figure 10. Reaction rate vs. space for radius = 1000 cm

Results for the different leakages are showed in Table 1. Percentage change in maximum reaction rate is calculated for different times at two days of interval. When percentage change is approximately equal, beginning of the steady state is considered. Calculation of FWHM and FW1M is done for 70th day. Velocity of wave propagation is calculated by taking ratio of (x/t), where x is the distance between the peak reaction rate at 70th day and the day when steady state begins (Transient Length) and t is the time difference between 70th day and the day when steady state begin (Transient Time).

Parametric studies of burnup wave for different values of radius are summarized in Table I.

TABLE I. RESULTS WITH DIFFERENT RADIUS

Radius (cm)	**TT (Days)**	**TL (cm)**	**FWHM * (cm)**	**FW1M * (cm)**	**Velocity (cm/year)**
R = 100	14	9.0	13.50	42.25	162.9
R = 150	16	11.0	13.50	43.00	202.8
R = 200	16	11.5	14.00	44.00	226.4
R = 400	16	11.5	14.25	44.00	227.0
R = 500	16	11.5	14.00	44.00	283.9
R = 600	16	11.5	14.00	44.00	290.6
R = 700	16	11.5	14.25	43.75	294.0
R = 800	16	11.5	14.00	44.00	294.0
R = 1000	16	11.5	14.00	44.00	300.8
R = 1500	16	11.5	14.00	44.00	300.8

*These parameter remain almost constant but maximum reaction rate keeps on decreasing as a function of time and space.

IV. CONCLUSIONS

From the results obtained above, it can be concluded that,

1. Leakage is too high for the case of smaller radius. Burnup wave is developed but it gets damped due to high leakage.
2. It can be seen from the variation of velocity of wave propagation, solitary burnup wave velocity is constant only when leakage is almost negligible.
3. Wave characteristics in terms of FWHM and FW1M indicates that reaction rate width is symmetric and FWHM and FW1M remain same as a function of radius of cylindrical medium (radius equal to 200 cm and above).
4. All the characteristics of solitary burnup wave is constant when leakage is zero.

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Introducing the FloatNuc Project: The KLT – 40S Floating Nuclear Power Plant Simulation at Various Sea State Condition

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I. Introduction

The KLT-40S floating nuclear power plant is first of a kind nuclear power plant in the world. The nuclear reactor is PWR type reactor and installed inside a floating craft. It will operate throughout it lifecycle on the sea. According to the previous research, the floating craft motions affected the thermal-hydraulics parameter and cause oscillation of parameters. Pitching, heaving, and rolling motions with high amplitude and low motion period affect the thermal hydraulic behavior of the reactor [1]. In this project, we want to know how is the effect of the floating craft motions in various sea state conditions to reactor dynamics parameters like the total reactivity. The total reactivity in the reactor system itself can conclude the behavior of the reactor dynamics. We also want to know the effects of coupling between two different motions, like pitching – heaving and rolling – heaving motions.

This project used ship keeping simulation to model floating craft motions in various sea state conditions. Coupling calculation between the ship keeping code, thermal-hydraulics code, and neutronic code is a hard approach and needs high computer specification. So, we made a new simulation software to simulate the nuclear power plant dynamics using a simplified method, the classical Runge-Kutta 4th order method. The simplified method was selected because we also intended to make a lightweight simulation for educating young engineers about floating nuclear reactor dynamics.

FloatNuc is a floating nuclear reactor dynamics simulation software to simulating KLT-40S floating nuclear power plant dynamics at various sea state conditions. The development of the software was using the C++ and Qt framework. We used the KLT-40S floating nuclear power plant specification as a basis in this simulation [2]. The initial value for the reactor dynamics simulation in FloatNuc software were obtained using SCALE 6.1 and the RELAP5-3D code [3].

II. FloatNuc Simulation Development

The current development of FloatNuc software is still developing the nuclear reactor dynamics for the reactor core. The simulation for primary loop is not yet developed due to the lack of verified data for the reactor pump and reactor pressurizer.

A. Nuclear Dynamics Equations

Equation (1) and (2), is the point dynamics equation and precursor equation to model the point nuclear reactor kinetic approach in this simulation [4], [5]. We use six group of delayed neutron in this equation. The point dynamics was selected, because we need low computational power method and need to simplify the energy of neutron into one energy approximation. Where $n(t)$ is neutron density, $\rho(t)$ is the total reactivity of the system, $\beta_{mix}(t)$ is delayed neutrons fraction, Λ is the prompt neutron generation time, $Q(t)$ is the external neutron source, $C_i(t)$ and λ_i are the number of delayed neutron precursors in group i and their decay constant.

The heat transfer between the reactor fuel to the reactor coolant also modeled in equation (3) and (4) using Newton's law of cooling with simplification and modification from this citation [6]. Where A_s is the heat transfer area, h is the convection coefficient, $\dot{m}$ is the mass flow, c_p is the specific heat capacity, T_F is the fuel temperature, T_M is the moderator temperature, ρ_F is the density of the fuel and ρ_M is the density of the moderator.

The reactivity feedbacks such as the fuel temperature feedback, the coolant temperature feedback, the nuclear poison feedback, the void feedback and the control rod worth were also modeled as the total reactivity in the equation (5). Where α_F is the fuel temperature coefficient, α_M is the moderator temperature coefficient, α_{Void} is the void coefficient, α_{Xenon} is the xenon coefficient, $\alpha_{samarium}$ is the samarium coefficient, $\rho_{ext}(t)$ is the external reactivity from control rod, $N_{samarium}$ is the number of samarium in the system, N_{Xenon} is the number of xenon in the system.

Equation (6) is determine the amount of fission power from the nuclear reactor. Where $P_{fission}$ is the fission power, Σ_f is the fission cross section, $vn(t)$ is the neutron flux, $E_{fission}$ is the fission energy, V_{core} is the volume of the reactor core.

$$\frac{dn(t)}{dt} = \frac{\rho(t) - \beta_{mix}(t)}{\Lambda} n(t) + \sum_{i=1}^{6} \lambda_i C_i + Q(t) \quad (1)$$

$$\frac{dC_i(t)}{dt} = \frac{\beta_i(t)}{\Lambda} n(t) - \lambda_i C_i(t), i = 1 \dots 6 \quad (2)$$

$$\rho_F V_F c_{p,F} \frac{dT_F}{dt} = P_{fission} - hA_s(T_F - T_M) \quad (3)$$

$$\rho_M\, V_M\, c_{p,M} \frac{dT_M}{dt} = hA_s(T_F - T_M) - \dot{m}c_{p,M}(T_{M,out} - T_{M,in}) \quad (4)$$

$$\rho(t) = \alpha_F(T_F(t) - T_F, 0) + \alpha_M(T_M(t) - T_M, 0) + \alpha_{Void}(\%Void) + \alpha_{Xenon}(N_{Xenon}) + \alpha_{samarium}(N_{samarium}) + \rho_{ext}(t) \quad (5)$$

$$P_{fission} = \Sigma_f\ \ vn(t)\ E_{fission}\ V_{core} \quad (6)$$

B. Sea State Conditions Code

The floating nuclear power plant will operate throughout its entire lifecycle on the sea. The sea state will change based on time and local weather conditions. The simulation model the operation of the nuclear reactor dynamic condition in ten different sea state conditions based on the World Meteorological Organization (WMO) sea state conditions code. The data of the wave heights, period and probability in Table 1 is perceived from northeren hemisphere annual sea state conditions [7].

TABLE I. NORTHEREN HEMISPHERE ANNUAL SEA STATE CONDITIONS

Sea State Condition	Parameters		
	Wave Heights (meter)	*Wave Period (s)*	*Probability (%)*
0	0	-	-
1	0 – 0.1	-	-
2	0.1 – 0.5	7	5.7
3	0.5 – 1.25	8	19.7
4	1.25 – 2.5	9	28.3
5	2.5 – 4	10	19.5
6	4 – 6	12	17.5
7	6 – 9	14	7.6
8	9 – 14	17	1.7
9	Over 14	20	0.1

We used Table 1 data to calculate the ship keeping characteristic of the "Akademik Lomonosov" floating craft using MaxSurf software. The further explanation of the calculation method can be seen in this citation [8]. The results in Table 2 show that the pitching degree of the floating craft can reach 9.16 degree with period of 6 second. Based on the previous research, the void will exist in this condition [1]. Table 3 show the results of rolling – heaving motion. According to the previous research in rolling motion the void will not exist, but there is a coupling between rolling and heaving motion in this case. The effects of two coupling motions to the reactor dynamics need to be determine as different motion type.

TABLE II. SHIPKEEPING CHARACTERISTICS OF AKADEMIK LOMONOSOV IN PITCHING – HEAVING MOTION

Sea State	Parameters					
	Wave Heights (meter)	*Wave Period (s)*	*Pitching Degree*	*Pitching Period (s)*	*Heaving*	*Heaving Period (s)*
0	0	-	0	0	0	0
1	0 – 0.1	-	0	1.2	0	1.2
2	0.1 – 0.5	7	0.03	0.6	0.04	0.6
3	0.5 – 1.25	8	0.69	6	0.243	6
4	1.25 – 2.5	9	2.13	6	0.702	6
5	2.5 – 4	10	5.98	6	3.38	6
6	4 – 6	12	5.52	6	5.83	6
7	6 – 9	14	6.60	6	7.17	6
8	9 – 14	17	7.41	6	9.16	6
9	Over 14	20	9.16	6	14.14	6

TABLE III. SHIPKEEPING CHARACTERISTICS OF AKADEMIK LOMONOSOV IN ROLLING – HEAVING MOTION

Sea State	Parameters					
	Wave Heights (meter)	*Wave Period (s)*	*Rolling Degree*	*Rolling Period (s)*	*Heaving*	*Heaving Period (s)*
0	0	-	0	0	0	0
1	0 – 0.1	-	0	0	0	0
2	0.1 – 0.5	7	0.25	8	0.20	8
3	0.5 – 1.25	8	0.55	8.91	0.29	8.91
4	1.25 – 2.5	9	5.69	9.43	0.87	9.43
5	2.5 – 4	10	7.69	10.43	2.90	10.43
6	4 – 6	12	20.44	11.15	5.13	11.15
7	6 – 9	14	14.90	11.29	6.03	11.29
8	9 – 14	17	11.25	11.40	8.14	11.40
9	Over 14	20	11.68	11.10	13.16	11.10

III. FLOATNUC SIMULATION SOFTWARE DEVELOPMENT AND RESULTS

A. Simulation User Interface

We were using the QT framework to build the modern C++ user interface and can be seen in Fig 1. This simulation provides the user with 17 different reactor parameters and three different simulation options. The user could change the option seamlessly and adjust the control rod position when the simulation is still running. The other feature is the user could see the floating craft animation and the control rod position during the simulation. The user could also simulate the insertion of reactivity in the core, the shutdown process of the nuclear reactor and the Loss of Flow Accident condition (LOFA).

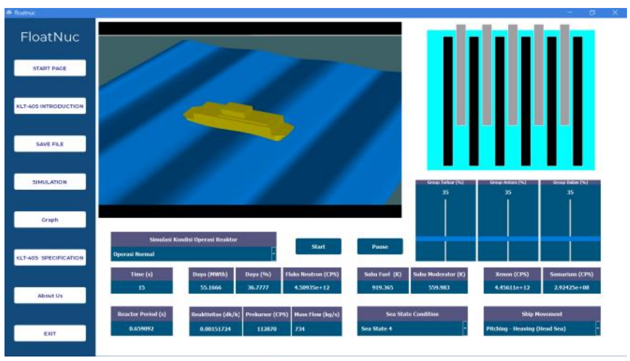

Figure 1. FloatNuc Software User Interface

B. *FloatNuc Simulation Result Verification with Reference and RELAP5-3D code*

The data of the floating nuclear power plant parameter are not only shown as a text but also as a graph between the value and the time. The number of parameters that could be visualized by FloatNuc software is 17 parameters. There is also a feature to save the result with .csv format so it can be analyzed using excel. In Table 2, we can see comparisons between the coolant temperature output from the RELAP5-3D code, FloatNuc software, and reference when the reactor power at 100% and sea state condition 0 [9].

TABLE IV. SIMULATION RESULT VERIFICATION

The Primary Coolant Output Temperature from FloatNuc Simulation (K)	**The Primary Coolant Output Temperature from Reference (K)**	**Error (%)**
571.49	589.15	2.99
The Primary Coolant Output Temperature from FloatNuc Simulation (K)	**The Primary Coolant Output Temperature from RELAP-5 (K)**	**Error (%)**
571.49	588.645	2.914

C. *Reactor Dynamics Simulation Result at Steady Conditions*

One of the important parameters in reactor dynamics is the total reactivity in the system. The total reactivity will represent the neutronic and thermal hydraulic condition of the reactor when experiencing oscillation effects from the floating craft motion.

In Fig 2, we can observe the oscillation of the total reactivity for various sea state conditions in the pitching - heaving motion. The total reactivity in the system begins experiences oscillations from sea state condition 3. The total reactivity amplitude is getting greater along with the increase of the sea state conditions. In the highest sea state condition, the total reactivity amplitude of the system is reaching 60 pcm. This value is quite large compared to the fuel temperature reactivity coefficient of -1.78pcm / K [2]. If we convert it into the fuel temperature, it equivalent to 33 kelvins.

According to equation (5), the total reactivity is the sum of reactivity feedback of the fuel temperature, coolant temperature, the presence of neutron poisons and the void formation in steady-state conditions. The total reactivity will affect the neutron balance, precursor balance and the heat transfer from the fuel into the coolant according to equation (1), (2), (3). (4) and (6). It also can indicate there is an oscillation in other reactor parameters, like reactor power, fuel temperature, void fraction and coolant temperature.

In Fig 3, we can perceive there isn't any oscillation of the total reactivity parameter for various sea state conditions in the rolling - heaving motion. It also can indicate there isn't any oscillation of the other reactor parameters. The variation of the total reactivity for various sea conditions in the rolling - heaving motion is occurred due to the void formation as sea state condition increases. Consequently, sea state condition 1 has the highest total reactivity and sea state condition 9 has the lowest one.

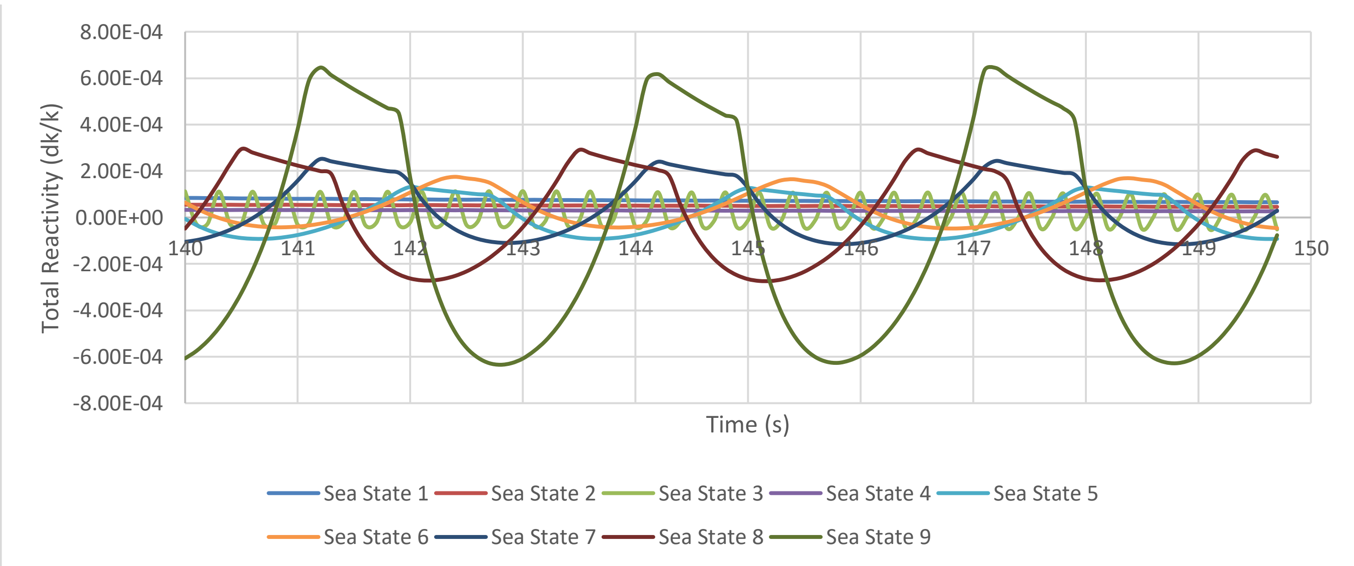

Figure 2. Oscillation of Total Reactivity Parameter in Pitching – Heaving Motions at Various Sea State Conditions

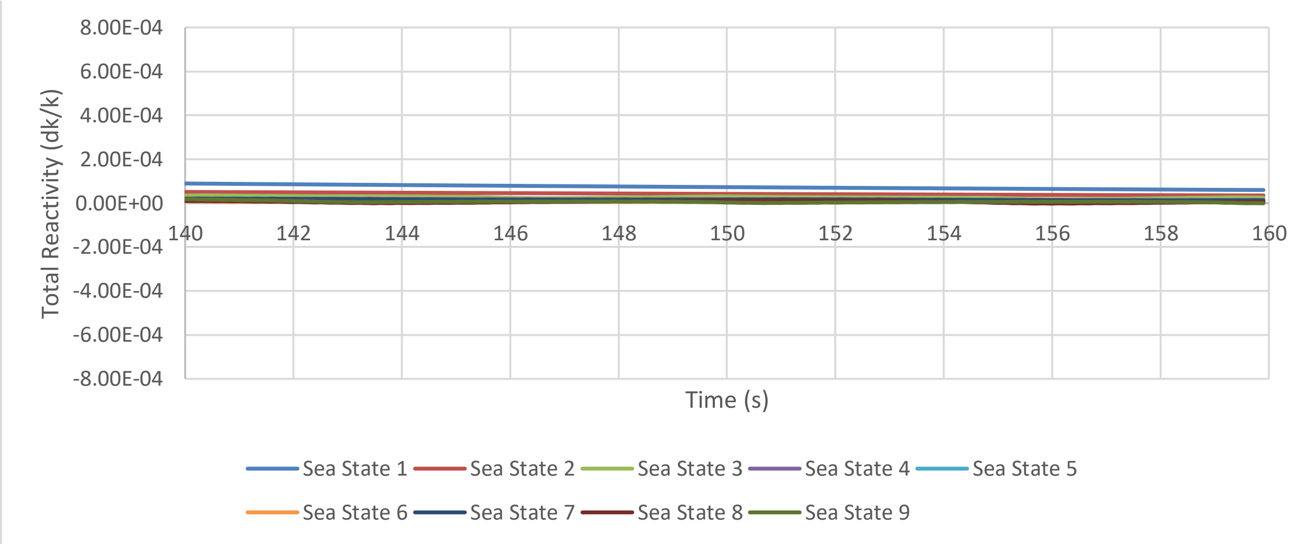

Figure 3. Oscillation of Total Reactivity Parameter in Rolling – Heaving Motions at Various Sea State Conditions

IV. FloatNuc Future Development and Challenges

For future development of the FloatNuc simulation project, we will develop the neutron dynamics equation in two different neutron energy, fast and thermal. We will also add the entire primary cycle of the nuclear reactor for better simulation and analysis. The addition of the reactor pressurizer, steam generator and the reactor pump will give a better approximation for the nuclear dynamics parameters value.

The addition of the nuclear power distribution and the void formations distribution calculations also planned, for better educational purposes in nuclear reactor dynamics behavior at various sea state conditions. The challenges according to the additional feature and analysis are in the computational power for this simulation software. Current development is still using the single core of the computer processor. Future challenges are how to add multi-core for the calculation process and the memory management for plotting and storing simulation data.

V. Conclusion

The FloatNuc simulation software can simulate the behavior of the KLT-40S floating nuclear power plant at various sea state conditions with less than 3% of error. According to the simulation, the oscillations of the total reactivity parameter is higher in pitching – heaving motion than the rolling – heaving motion. The highest amplitude of the total reactivity oscillations is examined in pitching-heaving motion with value reach 60 pcm at sea state condition 9. The simulation it self is still needing further development to be able provide detailed data about the KLT-40S reactor dynamics.

Acknowledgment

I would like to acknowledge the support provided by the Department of Nuclear Engineering and Engineering Physics, Universitas Gadjah Mada for this project.

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Preliminary Neutronic Design of High Burnup 15 Times Recirculation Cycle Pebble Bed Reactor

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I. INTRODUCTION

This research was carried out in the Center of Reactor Safety Technology (PTKRN BATAN), where there was a research project related to the development of the Electric and Steam Displacement Reactor for Industry (PLUIt) with a design profile that followed the HTR-PM namely the Chinese state-owned commercial power reactor with the High Temperature Gas Coolant type Reactor (HTGR). The project requires a lot of initial research regarding the safety and optimization factors of this reactor. This research is concerned with optimization of burnup discharge, fuel temperature after the occurrence of Depressurizer Loss Of Coolant Accident (DLOFA), power peaking factor and power distribution on the reactor core. In this research, a 15-multipass recirculation of fuel is analyzed. Operating Power used by this reactor uses 150 MWt less power than the HTR-PM reactor which is 250 MWt. This aims to get a small and compact reactor design. The parameters changed are terrace height and core radius by maintaining power density of 3.27 W / cc. From these various parameters, the most ideal combination will be chosen in determining the reactor geometry and fuel recirculation through neutronic analysis. The results of the analysis are expected to get a smaller core geometry by meeting the safety limit requirements.

II. METHODOLOGY

A. *Core Design and Pebble Bed Reactor Fuel Handling Reactor*

The PBR core scheme is shown in Figure 1. The reactor terrace is enclosed by a graphite reflector which also forms the core supporting structure. Between the graphite reflector is a control rod channel where there is a control rod that can move vertically according to operating requirements. In general, a PBR core can be controlled with the position of the control rod from outside the terrace as long as the diameter of the active terrace is not more than 3 m. There is a cold helium canal through which cold helium flows from the steam generating unit to the top of the terrace. Then helium will flow down on the reactor core to cool and take heat from the terrace. The hot helium, which has passed through the terrace, flows into the steam generating unit via a coaxial hot gas duct. Spherical pebble fuel with a diameter of 6 cm is located on the reactor core at random and moves axially during operation until it comes out of the bottom terrace. From the operational side, this fuel scheme allows refueling even though the reactor is operating. Neutronically, this ability makes the core reactivity able to be maintained stable [1].

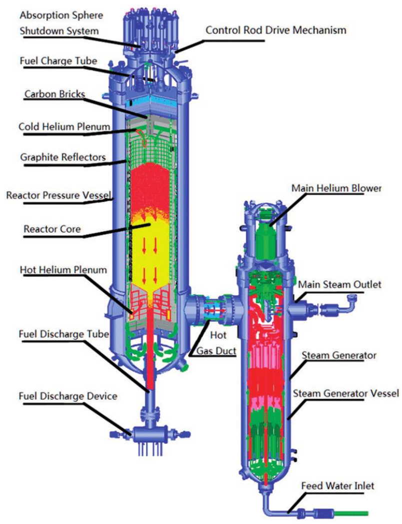

Figure 1 Schematic Cross Section of the HTR-PM Core View[2]

Pebble fuel handling in PBR reactors is shown in Figure 1. From the storage of fresh fuel, fuel is fed to the terrace by the distributor unit according to the design rate of refueling. On the PBR reactor core, the fuel moves axially down until it finally exits from the bottom of the terrace. After passing through the singulator unit, one by one each pebble fuel will experience a geometry damage check. If there is a geometry damage, the fuel will proceed to the storage of the damaged fuel. The fuel which is still intact then experiences a fuel fraction check level. As for PBR reactors that implement multi-pass recirculation, it will depend on the results of the measurement of the level of the fuel fraction. Pebble fuel that has passed a certain limit will be stored in used fuel storage, while pebble fuel that has not met the target fuel fraction will be returned to the reactor core. Fuel transportation is done by utilizing the pebble-shaped fuel feature that is easy to flow with the pull of gravity. It also utilizes gas pneumatic technology to carry out pebble transportation from higher parts[3].

B. Analysis Method

Calculation of equilibrium is done using PEBBED software [4]. The neutron energy spectrum is divided into 8 groups, 4 groups each for fast and thermal neutrons. PEBBED has been validated through comparison with VSOP software and is used intensively for PBR design including for deep burn applications using fuel from Trans-Uranium (TRU) [5]. PEBBED simultaneously solves the diffusion equation and the fuel depletion equation by paying attention to the axial movement of fuel during reactor operation.

Based on the design of HTR-PM as a reference for this study, the design changes the height and diameter of the reactor core starting from 1.5 m to 3 m by maintaining the core power density of 3.27 W / cc. The amount of fuel recirculation on the core remains from the initial value of 15 (HTR-PM design). Reactor power was reduced from 250 MWt to 150 MWt.

III. RESULT AND DISCUSSION

The relatively small terrace geometry design will make the reactor building a minimalist and economical one. Increased core geometry will increase power density and reduce burnup value and fuel residence time. High power density causes a decrease in the safety factor, while the burnup value and low fuel residence time indicate less than optimal fuel. This requires us to choose a relatively small geometry, but still pay attention to the values of the power density, burnup, and fuel residence time so that the design of the terrace that is designed still has a high optimization.

Another review to consider is regarding the safety aspects of the reactor, namely Peaking Factor and fuel temperature in a state of depressurization loss of forced cooling (DLOFC). Each of these safety parameters has certain thresholds, including: Average fuel power density 3.2 W / cc, and fuel temperature when DLOFC is maximum 1620 ℃.

A. Fuel Burnup

Table 1 Effect of H / D on Burnup Discharge

Height (cm)	Diameter (cm)	H/D	mean discharge burnup (MWd/kg-HM)
234.00	500	0.47	81.6
288.89	450	0.64	82.42
365.63	400	0.91	82.39
477.55	350	1.36	80.92
650.00	300	2.17	76.98
936.00	250	3.74	69.42
1,462.50	200	7.31	55.34

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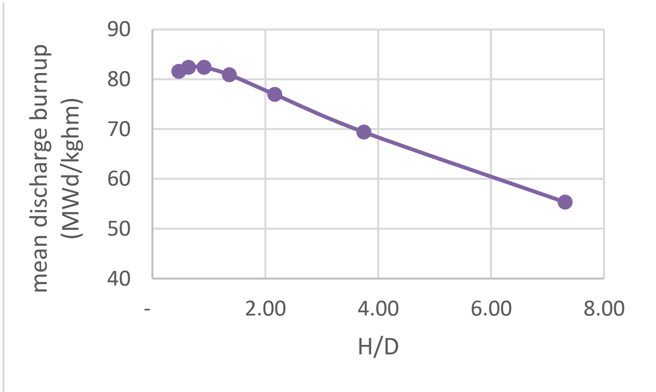

Figure 2 Graph of Discharge Burnup against H / D

From Figure 2, the greater the H / D, the smaller the burnup. However, at the beginning of the graph there is a burnup peak. This shows that there is an optimum point of H / D. Although there is an optimum point, the location of the point is at H / D between 0.47 to 0.64, which means that the diameter is around 4.5 m to 5 m greater than the height. In HTGR, the terrace diameter of 3 meters is the largest terrace diameter. If the core diameter is enlarged again, the heat inside the terrace will be difficult to flow out of the terrace so that the terrace design is even more complicated, namely the position of the control rod inserted into the reactor core. Otherwise in this study, the control rod is placed outside the reactor core. Therefore, this study only reviewed the diameter of the core not more than 3 m.

Table 2 Effect of core diameter on Discharge Burnup

Power Density (W/cc)	Height (cm)	Diameter (cm)	Discharge Burnup (MWd/Kg-HM
3.27	936	250	69.42
	848.97959	262.5	71.82
	773.55372	275	73.82
	707.75047	287.5	75.54
	650	300	76.98

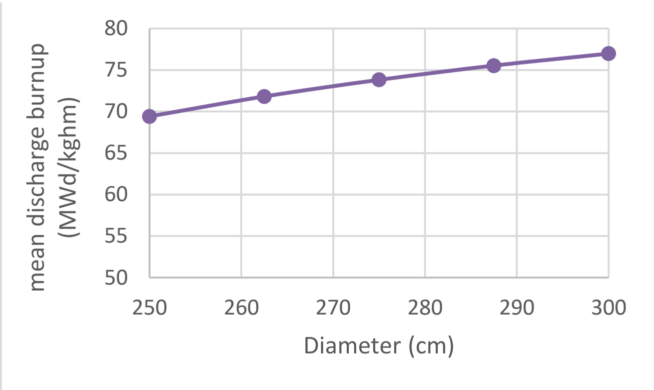

Figure 4 Graph of Discharge Burnup on core diameter

Seen from Figure 3, the greater the core radius, the higher the burnup. Changing the radius of the reactor core, the core height will adjust to keep the volume steady. In considering the economics of a building, a more minimalist building will be an option.

B. *Fuel Density Distribution*

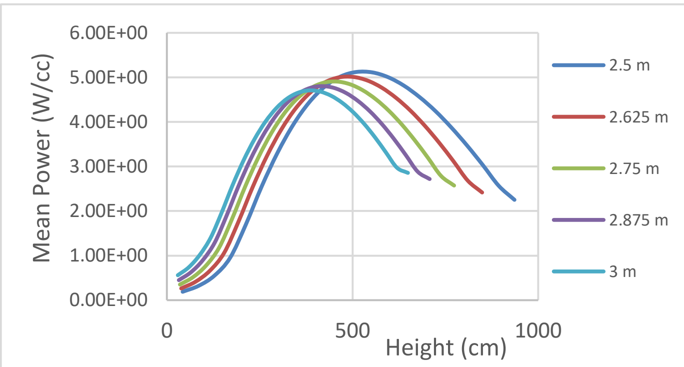

Figure 5 Graph of Axial Power Distribution with variations of core diameter (and height) at pass cycle 15

Seen from the comparison of Figure 4, in terms of diameter, the greater the diameter, the peak power density is getting to the left, which means the maximum power is at the axial lowest point. The range between the minimum power density and the maximum power density gets narrower as the diameter gets bigger. With these results, the power distribution looks more evenly distributed.

A good reactor core design has a fairly even distribution of power to support the safety factor. At more and more passes, the maximum power decreases which causes the reactor to be safer. In multipass 15, the difference between minimum and maximum power densities is not too wide and axial power is almost evenly distributed. From this, the number of cycles was chosen to be 15 times.

C. *Peaking Factor*

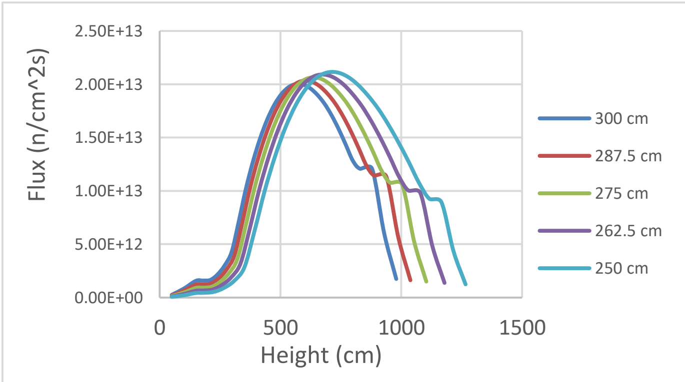

Figure 3 Graph of Thermal Flux Distribution with variations of core diameter at pass cycle 15

Table 3 Effect of core diameter on Peaking Factor

Power Density (W/cc)	**Tinggi**	**Diameter**	**Recirculation**	
			peak	Peak/mean
3.27	936	250	6.16	1.88
	849	262.5	6.09	1.86
	774	275	6.03	1.84
	708	287.5	5.97	1.83
	650	300	5.91	1.81

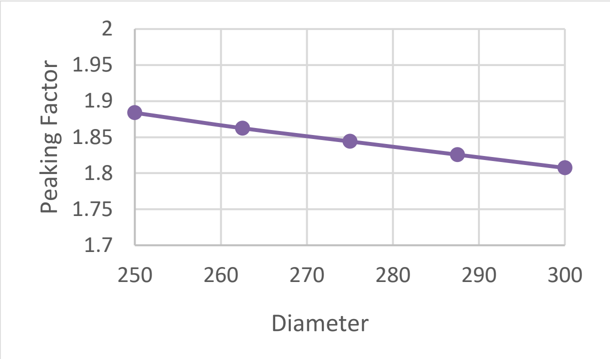

Figure 6 Peaking Factor graph of the core diameter

Peaking Factor is very closely related to power distribution on the core. From Figure 6, the average peaking factor is close to 2. The larger the diameter the more evenly distributed power distribution..

D. Fuel Residence Time

Table 4 Relationship between Pebble Speed and core diameter

Power Density (W/cc)	Height	Diameter	Pebble Speed (cm/day)
3.27	936	250	17.37
	848.97959	262.5	15.22
	773.55372	275	13.495
	707.75047	287.5	12.068
	650	300	10.876

Pebble speed is inversely proportional to its height. Pebble speed also has a big effect on fuel residence time. The higher the terrace the fuel residence time decreases.

Table 5 The relationship between Fuel Residence Time and core diameter

Power Density (W/cc)	Height	Diameter	*residence time (day)*
3.27	936	250	826.703
	848.97959	262.5	855.9899
	773.55372	275	879.866
	707.75047	287.5	900.5862
	650	300	918.0961

E. Fuel temperature when DLOFC

All safety aspects data, concludes that all geometry and pass variations tested meet the HTR-PM safety requirements with 150 MWt power except for OTTO cycles with 2.5 m, 2,625 m, and 3 m diameters not eligible because the fuel temperature exceeds 1620 ℃..

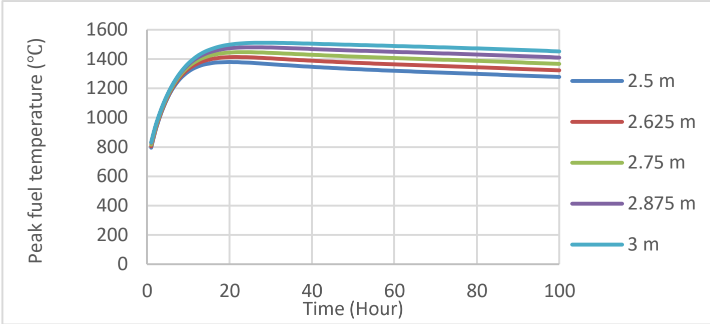

Figure 7 Graph of Fuel Temperature against time at Pass Cycle 15

IV. Conclusion

The conversion of HTR-PM power from 250 MWt to 150 MWt has been carried out. This change in power is also followed by changes in core geometry. Core geometry is varied by reviewing calculation parameters. From the data we have above, we can conclude that the 150 MWt HTGR has an optimal core design at a core radius of 1.5 meters which is the same as the 250 MWt HTR-PM, terrace height of 6.5 meters which was previously 11 meters. The calculation parameters of the HTGR core design are as follows. Burnup of 76.98 MWd / kg. Fuel residence time for 918 days or 2.52 years.

Acknowledgment

We would like to express our gratitude Dr. Hans Gougarat Idaho National Laboratory for the PEBBED Code

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An Artificial Neural Network Solution to the Point Reactor Kinetics Equations with One Precursor Group

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I. INTRODUCTION

Analysis of the point reactor kinetic (PRK) equations provides insights on reactor behavior during reactivity insertions. The PRK equations are a set of nonlinear ordinary differential equations (ODEs) given by:

$$\begin{cases} \dfrac{dN(t)}{dt} = \dfrac{\rho(t)-\beta}{\Lambda}N(t) + \Sigma_i^6 \lambda_i C_i(t) \\ \dfrac{dC_i(t)}{dt} = \dfrac{\beta_i}{\Lambda}N(t) - \lambda_i C_i(t) \end{cases}, \tag{1}$$

where $N(t)$ represents neutron density (neutrons/cm^3) inside the reactor core, $C_i(t)$ (1/cm^3) is the concentration of the i-th fraction of the delayed neutron precursors, β_i is a unitless parameter that represents the delayed neutron fraction of the i-th precursor group, β is total fraction of delayed neutron fractions, λ_i is the decay constant for the delayed i-th neutron (1/s), Λ is the mean neutron generation time (s), and $\rho(t)$ is the unitless reactivity function, which is a measure of the state of a reactor in comparison to where it would be if it were in a critical state.

The constants mentioned in the previous paragraph are treated as a given, making $N(t)$ and $C_i(t)$ the only unknowns that need to be solved. Therefore, (1) is a system of ODEs with 7 equations and 7 unknowns, namely $N(t)$ and $C_i(t)$, making this system of equations solvable.

Traditional numerical methods like Euler's Method or the 4th Order Runge-Kutta (RK4) Method are normally employed to solve (1) [1]. However, these methods mostly involve discretizing space into equally spaced points. These methods do not learn the solution online, but rather run as long as a certain error tolerance has not been met. These methods also do not generate an actual function, meaning that if a value outside the scope of the generated solution array is needed, interpolation would have to be employed if the value needed is within the domain, or regression if the value is outside of the domain. In either case, auxiliary methods had to be employed when using either Euler or RK4.

Another set of methods that work are neural networks (NNs). NNs are, by the Cybenko theorem, universal function approximators because they are composite functions that stack multiple systems of nonlinear equations, giving them the ability to model and generate complex patterns from data [2]. Neural networks also take a model-free approach that seeks to reduce the need for domain knowledge. The main advantage of using neural networks is that it automates the modelling of dynamical nonlinear systems just from data.

A. An Overview of Neural Networks

To simplify neural networks, we consider one input layer, one hidden layer network with ten nodes using sigmoid as its activation function, and one output layer. This is the architecture employed in this paper (see Fig. 1).

Mathematically, the output of the network would be the following:

$$NN = \sum_{i=1}^{H} v_i \sigma(z_i) + c_i \,. \tag{2}$$

In (2), $z_i = w_i x + b_i$, where w_i is the weight parameter from the input node to the i-th hidden layer, b_i is the bias term for the i-th hidden layer, and x is the input term, for which the output will be calculated. Similarly, v_i and c_i both represent the weight and bias terms respectively from the hidden layer to the output layer. The sigmoid activation function, σ, is defined as:

$$\sigma(t) = \frac{1}{1+e^{-t}} \tag{3}$$

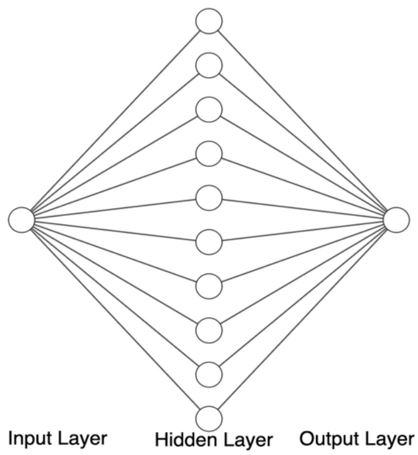

Figure 1: Neural Network Architecture

Mr. Joseph Bautista is grateful for the scholarship and financial support provided by the DOST-Science Education Institute.

In order for the NN to reach its optimal weight and bias term values to accurately model a function, a cost function has to be minimized which is typically the mean squared error. Several minimization techniques can be employed but the most commonly used is the gradient descent which is a variant of the first-order gradient-based optimization technique. This is because improvements over vanilla gradient descent like Momentum, Nesterov's Accelerated Gradient, RMSProp, AdaDelta, and Adam are better at reaching the global minima (assuming it exists) [3].

B. *Advantages of NNs*

NNs go away with auxiliary methods such as interpolation and regression as NNs directly generate a function by minimizing a defined cost function. Overfitting is also often a problem in NN research, but not for solving ODEs, since what the NN is solving for is a one-to-one correspondence. In traditional NN applications like regression, the data is not one-to-one, which makes overfitting a problem.

Other benefits of using NNs are as follows [4]:

1) Avoids the stiffness problem that traditional methods like RK4 and Euler encounter
2) Generates a closed analytic form of the solution
3) Less computational complexity as it generates both a compact data set and a solution model
4) Method is general and can be applied to any differential equation problem
5) Solution can be realized in hardware for real-time engineering applications

This is not the first time NNs are used to solve for the PRK equations [5], however, they used linear fractional neutron models, which is not a soft introduction. In this work, we introduce a basic application of using NNs to solve the PRK equations. In particular, our work allows for an introductory approach towards eventually extending the application to the standard six precursor groups version—a system of seven coupled nonlinear ODEs—and eventually, the neutron transport equation.

II. METHODOLOGY

A. *Neural Network Set-Up*

We define a trial solution $\psi_T(t)$ as:

$$\psi_T(t) = A + t \times NN(t, P) \tag{4}$$

where A is our boundary condition, *NN* is our neural network, and P is the weight and bias terms for *NN*. Since NNs are universal function approximators, the derivative of the trial solution can approximate the ODE at hand. Therefore, the minimization can be expressed as:

$$J(P) = \frac{1}{N}\sum_{i=1}^{N}\left\{\frac{d\psi_T(t_i)}{dt} - \frac{df\big(t_i, \psi_T(t_i)\big)}{dt}\right\}^2 \tag{5}$$

We define f here as the ODE being solved. On the other hand, J is defined as the cost function. The minimization procedure involves minimizing the cost function, which is the difference between the trial solution and the given ODE $\forall\ t_i$. As for the ODE, we plug in t_i and the current value of the trial solution at t_i. We update the weight and bias terms by taking the derivative of J with respect to the different weight and bias terms using programmed software [6] [7] [8] [9].

For systems of ODEs, the process is extended by creating one trial solution per ODE in the system, which means that our cost function J will be the sum of the different costs per trial solution and its opposing real solution:

$$J(P) = \sum_{k=1}^{M}\frac{1}{N}\sum_{i=1}^{N}\left\{\frac{d\psi_{Tk}(t_i)}{dt} - \frac{df\big(t_i, \psi_{Tk}(t_i)\big)}{dt}\right\}^2 \tag{6}$$

As was mentioned previously, the NN used for this research will have one input layer with one node, one hidden layer with ten nodes, and one output layer with one node. The activation function to be used is the sigmoid function, while the optimization technique utilized will be the Adaptive Moment Estimation (Adam) for faster convergence [10].

For training, the neural network was allowed to run for a maximum for 8000 iterations, but if the rounded value of the cost function for up to two decimal places equals zero, then the training would stop. Weight and bias parameters were initialized randomly under a normal distribution. Since the PRK have two ODEs, two NNs have be trained, each with associated weight and bias parameters. The initialized parameters for one NN were randomized separately from the initialized parameters for the other NN, which means that the convergence rates will naturally differ depending on their different initial values.

B. *Software and Hardware Utilized*

The program was run via a Jupyter notebook in Google Cloud Platform (GCP), utilizing a *c2-standard-16* virtual machine, which has 16 central processing units and 64 gigabytes of memory. The boot disk used was GCP's own *Deep Learning Base Image: Base m32*, a Debian based image with CUDA 10.0.

C. *Reactor Assumptions*

In order to compare RK4, Euler, and NN, Metha and Malhotra calculated the analytical solution to the PRK equations with one precursor group with the following parameters: $\beta = 0.0075$, $\Lambda = 6.0 \times 10^{-5}$ s, $\lambda = 0.08\ \mathrm{s}^{-1}$, $N_0 = 1$, and $C_0 = \frac{\beta N_0}{\lambda\Lambda}$ [11]. A positive reactivity $\rho(t) = 0.0015$ is considered which is five times less than the β value which could represent a reactivity insertion from a control rod withdrawal.

D. *Metrics and Discretizations*

We compared each method relative to their mean squared error (MSE) with the analytical solution provided in reference [11]. The one-group PRK equations were solved under the domain [0,1], which is discretized into 10, 100, and 1000 points. For easier reference, the discretization of the domain [0,1] into 10 points will be referred to as the "h=0.1 discretization scheme," since dividing the domain in 10 points results in a step size of 0.1. Likewise, "h=0.01 discretization scheme" and "h=0.001 discretization scheme" will represent the discretization of [0,1] under 100 and 1000 points, respectively.

The NNs capability to predict results outside of its trained dataset was also determined. This involves using an NN that is trained under one discretization scheme to predict values under different discretization scheme. As an example, the NN trained on h=0.1 was used to predict values under h=0.01 and h=0.001. This was done to demonstrate the flexibility of NNs to

III. Results and Discussion

Training the NN under different discretization schemes will stop if the combined cost function rounded off to two decimal places is equal to zero. This occurred only once, for the NN trained under the h=0.1 discretization scheme, which stopped the training at iteration number 6514. The NNs under the h=0.01 and h=0.001 schemes all went through the entire 8000 iterations.

Table I presents the effect of the stiffness problem on traditional numerical methods like Euler and RK4 when solving the PRK for the neutron density, $N(t)$. Predictions obtained with a step size of h=0.1 were grossly inaccurate, with Euler having an MSE of 7.243×10^{35} while RK4 had an MSE of 1.978×10^{96}. Nevertheless, significant improvements were observed in the results when the step size decreased to h=0.01 and h=0.001. The solution plots are shown in Fig. 2.

However, the NN results behaved differently. Stiffness was not a problem and though NNs trained under a specific discretization scheme were tasked to predict values under different schemes, the NNs still performed well. For instance, in Table I, an NN trained on the h=0.1 scheme gave an MSE of 4.312×10^{-6}, but when made to predict under the h=0.01 and h=0.001 schemes, the results were 6.990×10^{-4} and 7.052×10^{-4}, which are two orders of magnitude higher than the MSE in the h=0.1 scheme, but still low enough values to be accurate. These demonstrate that the closed analytic form that NNs generate can generalize well.

Similar results can be observed in Table II, where stiffness proved to be a problem for Euler and RK4 when used to solve the PRK equation for the precursor concentration, $C(t)$. Just like in Table I, the NN here also generalized well when tasked to generate values outside of its trained discretization scheme. The corresponding solution plots are presented in Fig. 3.

Not having stiffness as a problem and having a closed analytic form that generalizes well means that the NN solution may be more viable for real time applications. One immediate application in software engineering is the creation of reactor simulators with an NN as its backend. Based on the results, the generation of relatively accurate results can be met by training an NN once and saving its weight and bias values locally. It is computationally cheaper to store only the weight and bias values in an array when compared to saving big data structures containing the solutions of the equations per time step. Another approach to make it cheaper further would be to turn the NN into a low-cost web service by hosting the NN in a virtual machine in a cloud service—results can be queried via an application programming interface request. Open source frameworks such as *Tensorflow* and *PyTorch* are readily available to aid programmers from training NN models with graphical processing unit (GPU) support to web deployment.

TABLE I. MSE for numerical and analytical solutions to *N(T)*

N	Euler	RK4	NN Trained on h=0.1	NN Trained on h=0.01	NN Trained on h=0.001
h=0.1	7.122×10^{16}	1.170×10^{47}	4.312×10^{-6}	9.447×10^{-7}	1.830×10^{-6}
h=0.01	9.613×10^{-5}	3.468×10^{-7}	6.990×10^{-4}	2.598×10^{-6}	2.316×10^{-6}
h=0.001	5.969×10^{-7}	2.616×10^{-7}	7.052×10^{-4}	2.516×10^{-6}	2.201×10^{-6}

TABLE II. MSE for numerical and analytical solutions to *C(T)*

C	Euler	RK4	NN Trained on h=0.1	NN Trained on h=0.01	NN Trained on h=0.001
h=0.1	1.112×10^{47}	1.827×10^{47}	1.024×10^{-2}	9.258×10^{-2}	1.114×10^{-1}
h=0.01	1.119×10^{-1}	1.095×10^{-1}	5.975×10^{-3}	9.038×10^{-2}	1.095×10^{-1}
h=0.001	1.097×10^{-1}	1.095×10^{-1}	5.597×10^{-3}	9.076×10^{-2}	1.093×10^{-1}

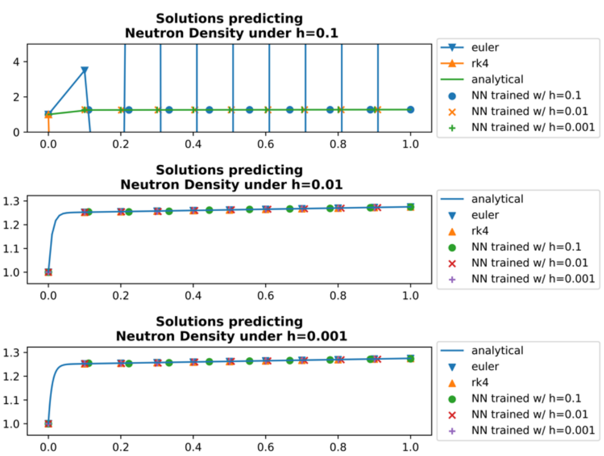

Figure 2: Comparison of numerical solutions for neutron density

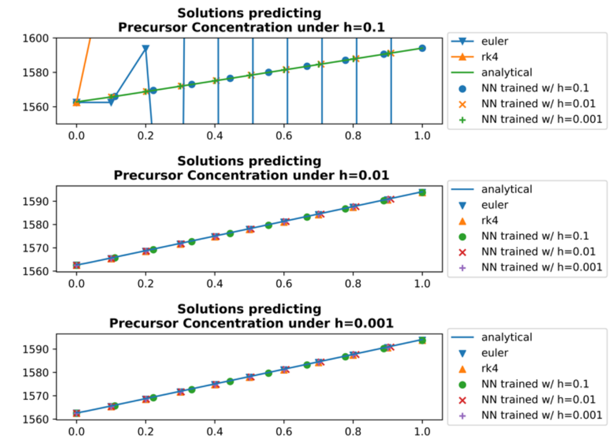

Figure 3: Comparison of numerical solutions for precursor concentration

IV. Conclusion and Further Work

The result of this work demonstrates that an NN-based solution provides a viable alternative to traditional methods like Euler and RK4. In particular, NNs are immune to stiffness and it can provide a solution in a closed analytic form that generalizes well along other schemes. This means that instead of using Euler or RK4 and using smaller step sizes, one could get away with a relatively bigger step size when creating a discretization scheme to feed into the neural network and still get good results. This implies that methods like interpolation and regression are not needed anymore if there is a need to generate a value outside of the step size range within the domain; the NN is robust enough to approximate those values.

Furthermore, NNs are viable for real time applications. Unlike Euler and RK4 that requires storing relatively larger data structures needed to store the solutions per time step, the only values that have to be stored for NNs in code aside from the architecture of the network itself is the value of the weights and bias parameters themselves, therefore, NNs will consume less computer space.

Future work that will be done to explore the flexibility of NNs include a more comprehensive comparison between NN and other traditional methods used to solve the PRK equations similar to the work done by Zhang, et al. [1]. Succeeding work will also focus on extending the current study to solve the six precursor group PRK, and comparing the NN-based predictions to either results from other papers that use traditional methods or actual reactor data. In the long term, it may also be viable to explore the application of the NN to solve the neutron transport equation.

With regards to the code, the utilization of a GPU for computations can be explored. The current work did not utilize a GPU because the Python libraries such as *NumPy* and *Autograd* could not use it natively, but deep learning frameworks like *PyTorch* and *Tensorflow* will have better computational performance as these two can naturally utilize the GPU. Other possible extensions of this work include varying the neural network's architecture, from exploring deeper networks, different optimization techniques, different activation functions, among others.

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Machine Learning Based Methodology for Assessment of Doppler Reactivity of Sodium Cooled Fast Reactor

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I. Introduction

Advanced fast reactors of the Generation IV represent new reactor concepts that are being studied to complement or replace current - day systems due to the important advantages they offer such as their capability to breed their own fuel from ^{238}U feed and to recycle actinides from their own spent fuel. Even though prototype reactors of this type are already built, these systems are still in the phase of the development and substantial research has yet to be done. In order to accelerate convergence of the fast systems multiphysics codes, as well as to provide operators of the already existing research reactors of this type with fast, yet accurate method of the core reactivity evaluation, considerable efforts are put in the development of the machine learning based methodology for the assessment of the core reactivity.

In the scope of this study, an attempt has been done to develop a new model for predicting the Doppler reactivity effect map for the Sodium - cooled Fast Reactor (SFR) core. The model is based on an artificial neural network (ANN) that takes 4 input parameters: two temperatures and two additional parameters characterizing different fuel compositions available in the core during the base irradiation and different possible sodium densities. These parameters are defined with the aim of enveloping all operational, as well as accidental regimes that could occur during the lifetime of the reactor core. Subsequently, artificial neural network is used to assess the 3D map of the Doppler reactivity for the specific core state, e.g. during normal operation or transients. The obtained results were compared to the results obtained by the utilization of the long established and widely used model based on the natural logarithm dependence of the core reactivity on the fuel temperature.

II. Parametrization

To develop a surrogate model for predicting 3D map of the Doppler reactivity for specific SFR core, 4 input parameters were selected: two fuel temperatures, spectral index (mainly to take into account possible sodium density variation) and fuel composition index (mainly to take into account fuel of different burnups).

A. Fuel temperature

The Doppler broadening of the cross - section resonances is particularly important phenomenon when safety of the nuclear energy systems is concerned. It improves reactor stability as it accounts for the dominant part of the negative fuel temperature feedback in thermal reactors and makes a substantial contribution in the fast reactors as well. This feedback coefficient is also called the prompt temperature coefficient, as it causes an immediate response to the changes in the fuel temperature. The prompt temperature coefficient of both thermal and fast reactors is negative, contributing to the inherent safety of the both systems.

It was determined empirically that the dependence of the Doppler reactivity on fuel temperature has the following form:

$$d\rho = \frac{K_D}{T} dT \qquad (1)$$

where ρ represents the reactivity of the core, K_D the Doppler constant and T the fuel temperature.

After integrating Eq. 1 from temperature T_o to temperature T, Eq. 2 is obtained:

$$\Delta\rho = K_D \cdot ln\frac{T}{T_0} \qquad (2)$$

B. Spectral index

For the purposes of this study, spectral index is defined as the parameter characterizing possible neutron energy spectra in different scenarios, including sodium voiding due to boiling or core overcooling. To be more precise, its purpose is to indicate which fraction of neutrons has energies in the resonance range, where Doppler effect is expected to prevail.

To account for impact of the coolant density on the neutron energy spectrum, spectral index (I_S) is defined as the ratio of the mass of the coolant (M_{cool}) to the mass of the fuel (M_{fuel}):

$$I_S = \frac{M_{cool}}{M_{fuel}} \quad (3)$$

A large spectral index corresponds to the enhancement of the neutron slowing down and therefore to the enhancement of the Doppler effect. On the other hand, a small spectral index (approaching zero as a minimum) corresponds to the sodium voiding and consequent spectrum hardening, therefore reducing the Doppler effect.

C. Composition index

While spectral index characterizes the intensity of the Doppler effect as a function of the coolant moderating power, a composition index characterizes the trade - off between positive and negative contribution to the Doppler effect. As Doppler effect causes broadening of the both capture and fission resonances, it could have both negative and positive effect on the reactivity, respectively. Subsequently, broadening of the capture resonances increases neutron capture on the fertile nuclei and reduces the reactivity of the observed system, while broadening of the fission resonances increases fission rate and increases the reactivity of the reactor core.

This effect is quantified by the composition index (I_C), defined as the ratio of the mass of the fissile fuel ($M_{fissile}$) to the mass of the fertile fuel ($M_{fertile}$):

$$I_C = \frac{M_{fissile}}{M_{fertile}} \quad (4)$$

Higher values of the composition index correspond to the higher amounts of fissile fuel and therefore less negative or even positive Doppler effect. On the other hand, lower values of the composition index correspond to the higher amounts of the fertile fuel, higher capture at fertile nuclei, and resulting more negative Doppler effect.

III. Methodology and Tools

A. Methodology

The basic idea of the herein proposed methodology is to develop an artificial neural network to be used for the calculation of the Doppler reactivity effect during transient calculations within the point reactor kinetics model. To that goal, an in - depth evaluation of all possible states of the SFR core is performed, according to which an extensive coverage of all of the expected values of the above defined parameters is executed. Even though database generation requires significant amount of the computational resources, its creation is expected to happen only once, after which it will be used to train neural network that will serve as a surrogate model.

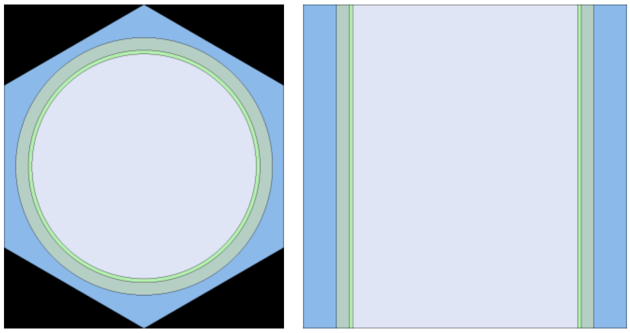

Figure 1. Radial and axial cross section of the unit cell.

B. Monte Carlo model

Both the core geometry and the fuel compositions for the current study are adopted from the European Sodium Cooled Fast Reactor (ESFR) core at the end of the equilibrium cycle [1]. Data base is generated via utilization of the Serpent 2 – a Continuous Energy Monte Carlo Reactor Physics Burnup Calculation Code [2].

In the early stages of the development of the methodology, two major simplifications are introduced: 1) fission products are not accounted for and 2) unit cell is used as a representative of the specific part of the core. According to 2), the system boundary conditions are set to ‘periodic’ in order to simulate the real world unit cell surroundings as accurate as possible. Run parameters of the simulation are set to obtain uncertainties on the order of 10 pcm. In addition, unresolved resonance data are used, as study deals with the fast spectrum systems. Geometry of the ESFR unit cell used to conduct systematic study of the core reactivity variation as a function of the fuel temperature, spectral and composition index can be seen in Fig. 1. More details can be found in [1].

The database for the training of the ANN was generated by running Monte Carlo simulations for unit cells with the predefined values of temperature (6 values), spectral index (11 values) and composition index (14 values). In total, 6×11×14 = 924 calculations were done.

More details on this could be found in the Table I, which represents an overview of the predefined parameter values used for the data base generation. These values are set in order to envelop all of the operational, as well as accidental regimes that could occur during the lifetime of the reactor core. For example, lower value of the spectral index correspond to the lower sodium density, present in the core during transients involving sodium boiling, while higher value correspond to the core overcooling or cold startup of the core. On the other hand, set of the lower values of the composition index (color coded: blue) corresponds to the fertile fuel of different burnups, while the set of the higher values (color coded: red) corresponds to the fissile fuel of the different burnups. Therefore, different values of the spectral and composition indices were obtained by varying sodium density and fissile - to - fertile fuel mass ratio, respectively.

Obtained 924 values of the k - effective represent enveloping 924 states of the reactor core. However, in order to assess the reactivity change as a function of the fuel temperature, already existing data base was transformed to account for all of the possible combinations of the fuel temperature change, including increase (e.g. 600 K → 1500 K), decrease (e.g. 1500 K → 600 K) or steady state (e.g. 600 K → 600 K), resulting in 6×6×11×14 = 5544 values of the reactivity effects, all of which are characterized by four input parameters: two temperatures, spectral and composition index.

C. Artificial neural networks

In the scope of this study, an ANN is trained to recognize the Doppler reactivity effect patterns within the four - dimensional space defined by the initial and the final fuel temperature, spectral and composition index and assess the reactivity change as a function of these parameters.

ANN training is performed by the utilization of the Neural Networks toolbox of the MATLAB. In particular, the MATLAB *fitnet* function is used for training the ANN (according to Levenberg - Marquardt optimization), as well as its testing and validation. This is performed by employing the 5544 train points described in the previous section. To be more precise, 70% of the input data were used for the actual training, while the remaining 30% were used for the validation and testing. Root mean squared error (RMSE), defined as the square root of the average squared difference between outputs and targets, is used as a measure of the quality of the ANN output. Due to the low computational intensity of the *fitnet* function, series of the neural network architectures were evaluated in search of the optimal configuration that will yield the lowest RMSE. ANN architecture resulting in the lowest RMSE (18 pcm) is shown in the Fig 2.

TABLE I. PARAMETERS OF THE ESFR UNIT CELLS USED IN GENERATING THE DATABASE FOR TRAINING OF ANN.

Temperature T [K]	**Spectral indices I_S [-]**	**Composition indices I_C [-]**
	0.00	
	0.05	0.0026 0.1150
300	0.10	0.0172 0.1198
600	0.15	0.0304 0.1244
900	0.20	0.0425 0.1283
1200	0.25	0.0525 0.1318
1500	0.35	0.0601 0.1353
1800	0.40	0.0688 0.1381
	0.45	
	0.50	

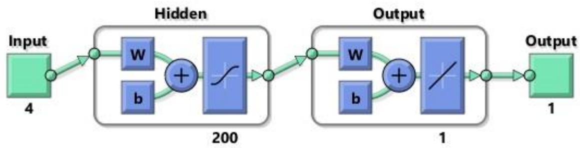

Figure 2. Neural network architecture.

IV. RESULTS

To evaluate the quality of the herein proposed machine learning based approach, it is compared to the traditional approach used for the calculation of the Doppler effect in the reactor point kinetics model based on the Eq. 2.

In order to assess performance of the developed ANN, the following 4 characteristic ESFR unit cells per fuel type (fissile and fertile) were evaluated using data from [1]:

- *Fresh flooded* unit cell: fresh fuel, nominal coolant density;
- *Fresh voided* unit cell: fresh fuel, zero coolant density;
- *Burnt flooded* unit cell: fuel at the beginning of the 6th cycle, nominal coolant density;
- *Burnt voided* fuel cell: fuel at the beginning of the 6th cycle, zero coolant density.

The Doppler reactivity effects were calculated for these 4 unit cells for both fissile and fertile fuels and for the two temperature intervals: from 300 K to 900 K and from 900 K to 1800 K. Therefore, in total 16 cases were evaluated using newly developed model (artificial neural network) and classical approach (evaluation of the Doppler constant by using natural logarithm function (Eq. 2)). Serpent 2 Monte Carlo calculations with statistical uncertainty on the order of 10 pcm were used as a reference.

These 16 cases enveloping previously defined regimes are simulated with the ANN by providing the corresponding values of the initial and the final fuel temperature, spectral and composition indices to it.

Regarding the logarithm dependence approach, values of the reactivity change are obtained by calculating Doppler constant on the basis of the two consecutive Serpent 2 runs for the highest (1800 K) and the lowest (300 K) fuel temperatures available for every unit cell. Subsequently, obtained Doppler constant is used to calculate the reactivity change for the temperature range of interest (300 K → 900 K and 900 K → 1800 K).

The results of the comparison are presented in the Table 2. As one can see, application of the new methodology allows for the **reduction of the relative error by a factor of 2** compared to the classical approach.

TABLE II. RESULTS FOR THE DOPPLER REACTIVITY EFFECTS OBTAINED BY SERPENT 2 (STATISTICAL ERROR ON THE ORDER OF 10 PCM), CLASSICAL APPROACH BASED ON THE NATURAL LOGARITHM DEPENDENCE (EQ. 2) AND THE NEWLY DEVELOPED ANN MODEL.

Fuel		Composition Index	Sodium Density	Spectral Index	Serpent 2				Eq. 2				ANN			
					Δρ [pcm]			KD [pcm]	Δρ [pcm]		ε [%]		Δρ [pcm]		ε [%]	
Type	*Burnup*	-	g/cm3	-	300 → 900	900 → 1800	300 → 1800	300 → 1800	300 → 900	900 → 1800	300 → 900	900 → 1800	300 → 900	900 → 1800	300 → 900	900 → 1800
Fissile	Fresh	0.1152	0.90355	0.12763	-906	-447	-1353	-755	-829	-575	-8%	29%	-890	-400	-2%	-10%
Fissile	Fresh	0.1152	0	0	-704	-397	-1101	-614	-675	-468	-4%	18%	-720	-380	2%	-4%
Fissile	Burnt	0.1379	0.90355	0.12763	-836	-399	-1235	-689	-757	-525	-9%	31%	-850	-370	2%	-7%
Fissile	Burnt	0.1379	0	0	-654	-353	-1007	-562	-617	-428	-6%	21%	-670	-330	2%	-7%
Fertile	Fresh	0.0028	0.90355	0.12763	-8385	-4769	-13154	-7341	-8065	-5590	-4%	17%	-8220	-4660	-2%	-2%
Fertile	Fresh	0.0028	0	0	-6621	-3812	-10433	-5823	-6397	-4434	-3%	16%	-6720	-3740	1%	-2%
Fertile	Burnt	0.0686	0.90355	0.12763	-2607	-1551	-4158	-2321	-2550	-1767	-2%	14%	-2630	-1530	1%	-1%
Fertile	Burnt	0.0686	0	0	-2248	-1144	-3392	-1893	-2080	-1442	-7%	26%	-2230	-1210	-1%	6%

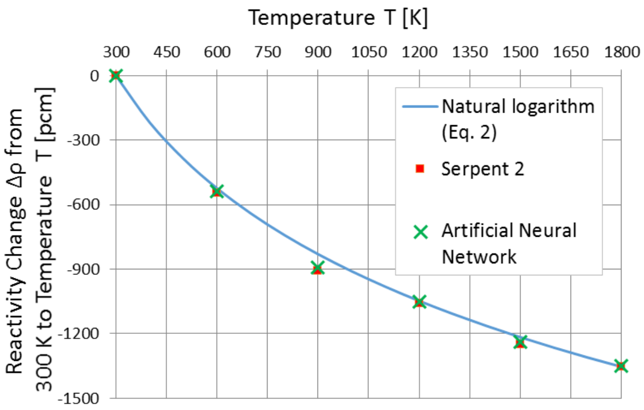

Figure 3. Comparison of the new model performance compared to the classical approach.

Moreover, the proposed approach allows for the **quantification of the change of the Doppler effect caused by the different fuel composition and the sodium density**, while classical approach given by the Eq. 2 includes only temperature dependence and additional modelling is necessary in order to introduce the dependence of the Doppler constant on the sodium density and the fuel composition.

For graphical illustration, the new approach is compared to the classical one in the Fig. 3., together with the overview of the ANN performance for the remaining available fuel temperatures.

V. CONCLUSIONS AND FUTURE WORK

To conclude, herein proposed approach for the calculation of the Doppler reactivity in conjunction with the point kinetics model shows a better performance compared to the classical natural logarithm approach.

The methodology presented in the scope of this study is just the core of the idea that offers a lot of space for the improvement, with potentially bright future. Some of the aspects of this methodology that should be addressed in the follow - up projects are inclusion of the fission products and increase of the coarseness of the variables space.

When fission products are concerned, authors propose introduction of the additional parameter that will take into account existence of the fission products thus making it an accurate representative of the fuel burnup. However, composition index is already partially accounting for the burnup characterization of the core, in particular through taking into account accumulation of the minor actinides with burnup. On the other hand, increase of the coarseness of the variables space has a simple idea of checking how far application of the neural networks could go when reducing the computational cost of the database generation without significant deterioration of the obtained results is concerned.

Additionally, after previously mentioned aspects are addressed, authors see the advancement of the methodology in the generalization of the variables space itself, as well as its boundaries, to attempt extending its applicability to the other fast spectrum nuclear energy systems.

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TRACK 4: THERMAL-HYDRAULICS

DIGITAL TWIN OF REACTOR UNIT AND ITS APPLICATION FOR DYNAMICS

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PIN-BY-PIN TRANSIENT SIMULATION OF THE “CONTROL ROD EJECTION” ACCIDENT FOR VVER-1000 IN THE KORSAR/GP CODE

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REWETTING ANALYSIS OF A NUCLEAR FUEL PLATE USING AN IMPROVED LUMPED MODEL

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NON-HYPERBOLICITY OF RELAP5 TWO FLUID MODEL UNDER LOSS OF FLOW ACCIDENT FOR A BOILING WATER REACTOR

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CFD SIMULATIONS OF THE OPAL COLD NEUTRON SOURCE

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Digital Twin of Reactor Unit and Its Application for Dynamics

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I. Introduction

The term “digital twin” came into common use around twenty years ago, and for the first time it was used in the aviation industry [1]. At present, this notion implies that, a computer model of a facility can be created based on its design data and this model can produce a fully-functional digital image. However, digital twins of complicated process facilities are still rather rare.

The digital twin of a reactor plant has a special value, since it reflects the performance of the reactor plant physical prototype and makes it possible to evaluate the facility current condition and to analyze the facility behavior in emergency modes and during limiting faults.

This paper presents the stages of creating a digital double for a reactor unit on the example of a nuclear icebreaker and NPP with PWR reactors

II. Digital Twin Composition

At present, with reference to a reactor plant, a virtual (digital) twin of a facility under study represents a composition of:

- software which is a unified system of calculation codes integrated into a common system environment and providing coordinated calculation of various physical processes and equipment;
- a system of verified input data (sets of model input data) to describe equipment behavior in design and emergency operation modes of the facility;
- hardware equipment enabling to carry out high-performance computing of mathematical models at high level of detalization.

At present Joint Stock Company (JSV) “Afrikantov OKBM” Rosatom is developing a unified system of calculation codes integrated into the common system environment and providing coordinated calculation of various physical processes and equipment. The system includes:

- one-dimensional code enabling simulation of reactor plant transient processes. The code uses a system of equations describing the reactor kinetics, heat and mass transfer processes, electromechanical processes in an electricity generating plant and control system operation algorithms. The package is aimed at solving problems related to water coolant (one-phase and two-phase processes, critical flow), noncondensable gases (air, nitrogen, hydrogen and helium) and metals (sodium), for example TIGR-1 program code is intended for joined neutronic and thermal-hydraulic analysis of non-LOCA transient. TIGR-1 code is developed by Rosatom. It is certified by Russian Federation Scientific and Engineering Center for Nuclear and Radiation Safety;
- the system of model-based designing of the functional software of automatic control systems where it is possible to create control algorithms, control interfaces and visualization of calculation schemes of mathematical models of various objects under study.
- a CFD code, coupled with the one-dimensional code, enables users to take into account the slight effects of non-isothermal coolant flows.

Figure 1 provides a flow chart of the unified system of calculation codes integrated into a common system environment and providing coordinated calculation of various physical processes and equipment.

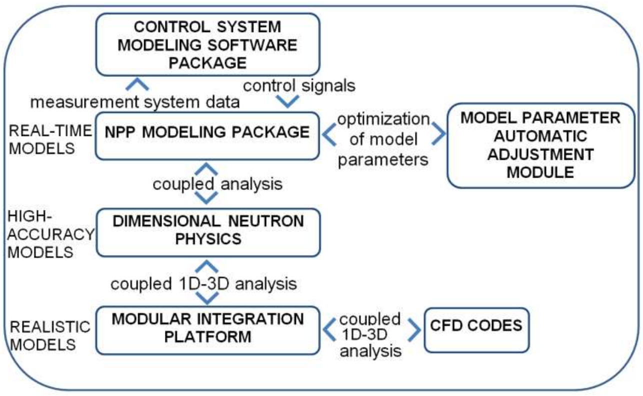

Figure 1. Digital twin development tools.

Such unified system provides the possibility of taking into account as 3-D effects of stratification of temperature flows in the pressure and discharge chambers of the reactor, as the possibility of calculating in real time-scale for training of icebreaker operators.

The common system environment allows to carry out related calculations of hydrodynamics, neutron physics, thermal-hydraulics and strength calculations of the strength and service life of the equipment, as well as to use a single model for all calculations.

The system of verified input data includes the database of all Project technical documentation developed from the design to putting into operation and a set of verification data obtained from the tests performed during commissioning operations, for example real pump characteristics, loop pressure drop, etc.

The hardware of the computational package consists of:

- a tool server for storage of mathematical models and the database under the Project;
- workstations where a mathematical model is being developed, locally debugged and run, and where the calculation process is being controlled;
- a system of interactive operating panels simulating real control panels of entire reactor and a system of displaying the calculation parameters.

Mathematical models, which can be created at any stage of facility's lifecycle and modified if required, constitute the basis of the digital twins under development.

Based on the experience of application, a digital twin or models (which constitute its basis) are required to validate the normal operation modes, determine the requirements for the control algorithms (based on the condition of ensuring the required quality of the transients), validate the modes with equipment failures, justify the sufficiency of the provided measures aimed at preventing the possibility of an emergency situation (due to the activation of preventive protection or an emergency reduction of power), justify the emergency modes by calculations ensuring conservatism of results from the point of view of safety criteria or justify the application of statistical methods of realistic analysis with conservative choice of equipment operation conditions.

At the stage of manufacturing and supply of the integrated control system of hardware, the digital twin is applied to correct the characteristic drawbacks of the existing process aimed at creating a functional software for the integrated control system of hardware. This increases the development and debugging time of the algorithms as well as improve the overall quality of the finished integrated control system of hardware.

At the stage of operation, the digital twin is used as the basis for the system of on-line parametric technical diagnostics of the overall state of the facility and the state of its individual components [2]. The need to use such kind of systems is due to the fact that the technical facilities become more sophisticated, their operating life grows and the automation degree of the control processes increases. Here great importance is attached to determining the facility condition which is being changed due to the impact of internal and external factors in the course of time. The technical diagnostics resolves issues related to determining the condition of technical facilities and pattern of their changes over time, which is aimed at determining the current condition of the facility to be diagnosed. The advanced technical diagnostics systems shall provide not only qualitative but also quantitative assessment of the condition of the facility under diagnostics [3]. This is ensured by the use of the digital twins of the facilities under study. Because it allows you to predict the state of the equipment not only in normal operation, but also in limiting faults, while reducing the conservatism inherent in safety analyses.

Further on, the obtained data are used for strength analyses, when justifying the residual life and during diagnostics of the actual state of equipment.

III. Conclusion

The operational data integrated in the digital twin make it possible to configure the computational models of individual equipment and of a reactor plant at large for their maximum compliance with the actual facility at all the stages of its lifecycle.

In JSC "Afrikantov OKBM" a computational environment is being developed including 1D and 3D thermal-hydraulic computer codes, codes for neutronic analyses, and software tools for creating control system models. This will allow modeling of fully-functional twins of reactor plants in the nearest future.

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Pin-by-pin transient simulation of the "Control rod ejection" accident for VVER-1000 in the KORSAR/GP code

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INTRODUCTION

In the safety assessment of VVER reactors a current important task is the reduction of the in applied safety margins. For example, in the case of increasing core power without changing core parameters, the existing safety margin is reduced. In this paper, research on the use of pin-by-pin modeling to resolve this issue described.

Usually the safety margins are calculated with use of the "hot channel" model. The fuel rods of the core with the highest power are combined into a separate hydraulic element (called "channel" in KORSAR/GP [1]) in this model. The power of the "hot channel" is set conservatively taking into account the axial non-uniformity coefficients (Kr, pin radial peaking factor). This channel includes the coolant that cools the fuel rods. The influence of the rest coolant in the core is not taken into account. This approach is conservative.

More realistic pin-by-pin modeling of a fuel assembly provides less conservative results that can be promising for nuclear power plant safety analysis. The pin-by-pin model gives the distribution of the fuel rods parameters (fuel and cladding temperature, fuel enthalpy, departure from nucleate boiling ratio) in the fuel assembly. This detailed data can be used for comparison with the acceptance criteria in safety analysis.

RESEARCH OBJECTIVE

The aim of this work was to assess the pin-by-pin model of the fuel assembly in the KORSAR/GP code. Because such a model contains a large number of elements (pin-by-pin model contains more elements then the whole model of reactor plant), which requires a lot of time for calculation, it is necessary to check the need for its application for different accidents. The application of this model was carried out for two accidents: steam line break upstream of steam isolated valve and the control rod ejection [2]. If we look onto the redistribution of the power distribution inside the fuel assembly during the transient, the control rod ejection accident is more representative. Therefore, in this work, the following tasks were set for this accident:

1) carry out the coupled neutron-physics and thermo-hydraulic transient calculation of control rod ejection with use of the fuel assembly pin-by-pin model;

2) compare the results with the simulation of transient results using "hot channel" model and estimate the highest possible power maximum obtained during the accident.

PIN-BY-PIN TRANSIENT SIMULATION BY KORSAR/GP

Pin-by-pin simulation approach in the code KORSAR/GP was proposed in the "A.P. Aleksandrov Scientific Research Technological Institute". This approach includes three stages for transient simulation:

- Simulation of the transient using full nodalization of the reactor plant. Nodalization includes one hydraulic channel for each fuel assembly. The macroscopic sections are also prepared for each fuel assembly. Neutron physics mesh includes 24 calculation nodes in the FA section which is the finest mesh for current version of KORSAR/GP. The fuel assembly is chosen where the worst conditions of fuel rods cooling are expected. The set of fuel assembly parameters is saved during the transient including fuel assembly inlet and outlet coolant pressure temperature and 2-group neutron fluxes for 24 calculation node for each calculation level. These parameters are then used as boundary conditions.
- Reconstruction of the pin-by-pin power distribution for chosen fuel assembly with use of special program RcMcFlux. This program uses dynamic neutron fluxes defined at the previous stage and steady state pin-by-pin power distribution. The dynamic pin-by-pin power distribution is saved and used as boundary condition at the final stage.
- Thermal-hydraulic pin-by-pin simulation of the chosen fuel assembly with use of boundary conditions obtained at stages number one and two.

In this paper this approach was applied for "Control rod ejection" accident.

DESCRIPTION OF THE ACCIDENT "CONTROL ROD EJECTION"

The "Control rod ejection" accident is characterized by a rapid injection of positive reactivity and a local power increase around the position of the ejected control rod. The initial state for the reactor system is 100% power. All initial values of thermal-hydraulic parameters were set to nominal values not considering uncertainties. The beginning of the stationary fuel loading campaign is selected. At the beginning of the cycle, the reactivity coefficient for the fuel temperature has its lowest absolute value, and therefore the negative feedback action on the fuel temperature will be minimal. The ejection is understood as the rapid (0,1 s) movement of the control rod from the bottom of the core. As a result, reactivity is introduced in a short period of time, which leads to an increase in the neutron flux and power redistribution over the whole core volume. There is a power redistribution in the core with a maximum in the area of ejected CR (control rod).

The purpose of the calculations in safety analyses is the verification of compliance with the acceptance criteria. The evaluation of the criteria parameters is performed using the "hot channel" model. "Hot channel" nodalization is presented on the Fig. 1. It consists of one heat structure and one hydraulic element (channel). All fuel rods of the one FA are modeled by a single heat structure element (HCS). Before the calculation several heat loaded fuel assemblies are selected, for which the relevant parameters will be evaluated. The nodalization of the "hot channel" model is similar to the nodalization of thermohydraulic channels simulating FA in the core, but is not connected hydraulically with the nodalization of the reactor plant. For the "hot channel" as the boundary conditions are set the values of pressures and temperatures obtained at the inlet and outlet of the selected fuel assemblies when calculating the transient. The power of the "hot channels" is set conservatively taking into account the axial non-uniformity coefficients (Kr, pin radial peaking factor). The axial non-uniformity is taken into account by setting the Kr – coefficient of fuel rod power distribution in the core (pin radial peaking factor).

It is usually assumed that pin-by-pin power distribution in the FA is not changing during the accident and the FA power increases in proportion to the power of the reactor plant. However, in the accident with control rod ejection power redistribution in the FA is possible. Therefore, even more conservative values of Kr coefficients are set for this accident. Stationary-state calculations are used to select the "hot channels" and the Kr coefficients, in which the working group is at the bottom of the core, and the accident FA is at the top. A cartogram of these conditions is shown in Fig. 2.

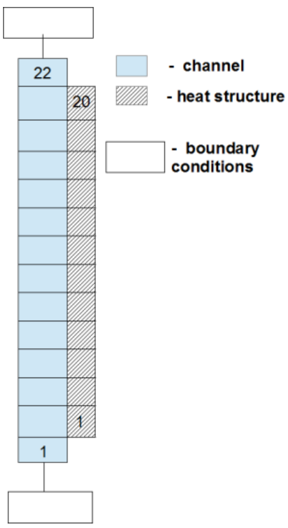

Figure 1. The nodalization of "hot channel" model

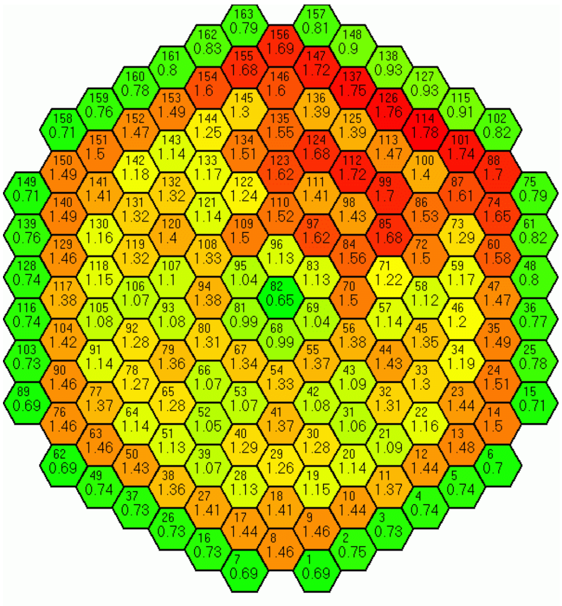

Figure 2. Distribution of Kr in the core for steady-state with insert working group and ejected control rod

The maximum Kr coefficient is set conservatively in the core. As figure 2 shows, the largest Kr in the stationary state is achieved in FA № 114. Setting such large coefficients replaces the lack of knowledge about the power distribution FA during the accident.

Only the first few seconds of the accident are considered in this report because the maximum values of the criteria parameters are achieved in this period. At the point of control rod ejection, a sharp increase in power occurs in the locality of the ejected control rod. This increase in power is limited by the action of negative feedbacks. Due to the actions of feedbacks on the fuel temperature, the compensation of the introduced reactivity takes place.

Description of the pin-by-pin model nodalization

The pin-by-pin thermal-hydraulic model of the VVER-1000 fuel assembly was developed with the code KORSAR/GP.

The nodalization of the developed model consists of parallel 1D hydraulic elements ("channels"). Each fuel rod is modeled by a separate heat structure element (HCS). The pin-by-pin FA model contains 312 fuel rods, 660 thermo-hydraulic channels, 18 guide tubes and 1 instrumentation tube (as fuel rods with zero power). The cross-section example of the fuel assembly pin-by-pin model is shown on the Fig. 3 (cross-section is presented for example, the pin-by-pin model was developed for off-centre instrumentation tube). The circles on the figure indicate the elements HCS: dark color – is for fuel rods, light color – is for control rods and guide channels. The coolant mixing between different FAs is not modeled. Thus the influence of the coolant outside the investigated FA is not taken into account.

The pin-by-pin model gives the distribution of the fuel rods parameters (fuel and cladding temperature, fuel enthalpy, departure from nucleate boiling ratio) in the FA.

This detailed data can be used for comparison with the acceptance criteria in safety analyses.

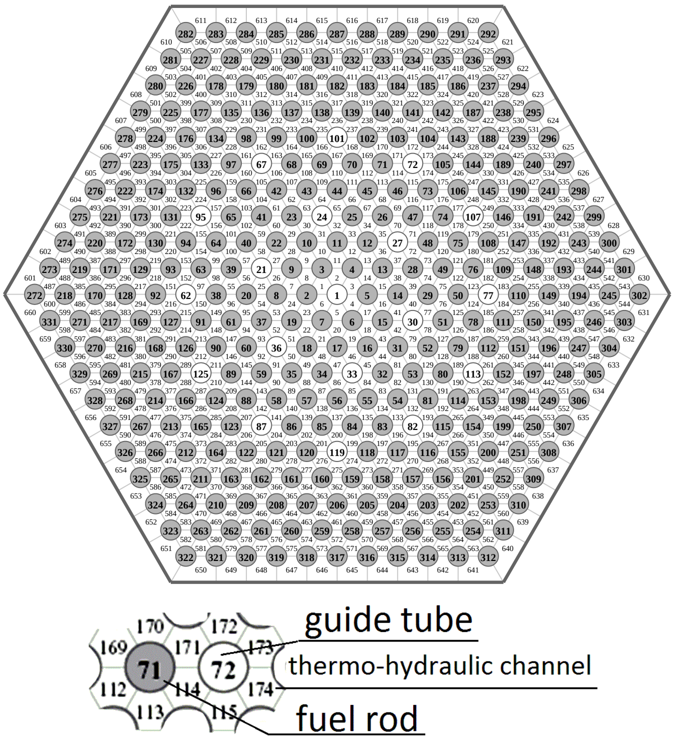

Figure 3. The cross-section example of the fuel assembly pin-by-pin model (numbers of the different elements)

The developed nodalization was verified to check of the pin-by-pin model correctness in a simple test. The test is focused on the flow rates in two models (pin-by-pin model and “hot channel” model) with the same FA inlet and outlet boundary conditions (pressure and temperature). The power of the heat structure was not specified. The flow rate difference of the two models is not higher than 3 %. Thus, the pin-by-pin model is correct.

The new KORSAR/GP pin-by-pin model was cross-validated with the TIGR-SP [3] code which is used by OKB “GIDROPRESS” for FA steady-state operation assessment. DNBR (departure from nucleate boiling ratio) difference between KORSAR/GP and TIGR-SP does not exceed 3%. Thus, the KORSAR/GP pin-by-pin model provides correct results.

The neutron-physics model was cross-validated with the code SAPFIR_95&RC_VVER [4] which is used for obtaining steady state pin-by-pin power distribution and power distribution of the fuel assemblies in the core. In the frame of this paper pin-by-pin power reconstruction methodology correction was indicated a small deviation of the relative power for fuel rods.

Calculation Results of “Control Rod Ejection” Accident Using Pin-by-Pin Model

The results obtained for the most heat loaded fuel assemblies during the accident were presented in the Table I.

We see that the maximum power of a pin is realized in FA № 99 and is 260 kW. When calculating separately FA № 99 using boundary conditions, we obtain a less conservative result. Figure 4 shows the results of calculation of the maximum pin power in FA № 99 using different models. However, after calculation of different fuel assemblies using the pin-by-pin model the following results were obtained (Table I).

TABLE I. Pin power maximum, kW

FA №	Hot channel model	Pin-by-pin model
99	260	208
113	223	178
114	250	217
124	259	210
125	212	170
126	245	215
137	245	214

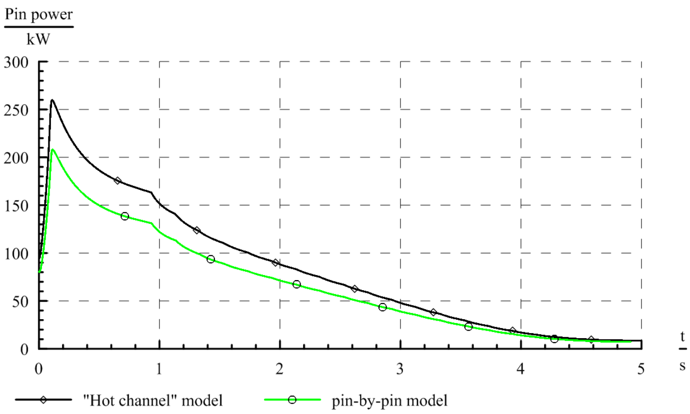

Figure 4. The maximum fuel rod power in the FA №99

The maximum pin power is observed in FA №114. It should be noted that the FA capacity is fully consistent with the 3D calculation. Such results can be explained as follows. In transient simulations, feedbacks on the average values of the parameters are taken in the FA area into account.

The value of the power maximum is determined by the value of the introduced reactivity and the action of feedbacks. The initially “colder” was the pin and coolant around it, the stronger will be the feedbacks in fuel temperature and in coolant density. In the stationary state, the power distribution field in the concerned area is as follows (Fig.5).

It can be seen that the insertion of the working group led to the distortion of the field in the FA № 112 and in this area the lowest values of power are observed. As a result of the control rod ejection a significant power field redistribution within the studied fuel assemblies occurs. Thus, for “hot” fuel rods, whose fuel temperature is higher than the average fuel cross-section value, which is used to calculate the reactivity effects, we will obtain a conservative power value. In opposite, for “cold” fuel rods with lower fuel temperature, the power distribution will be underestimated.

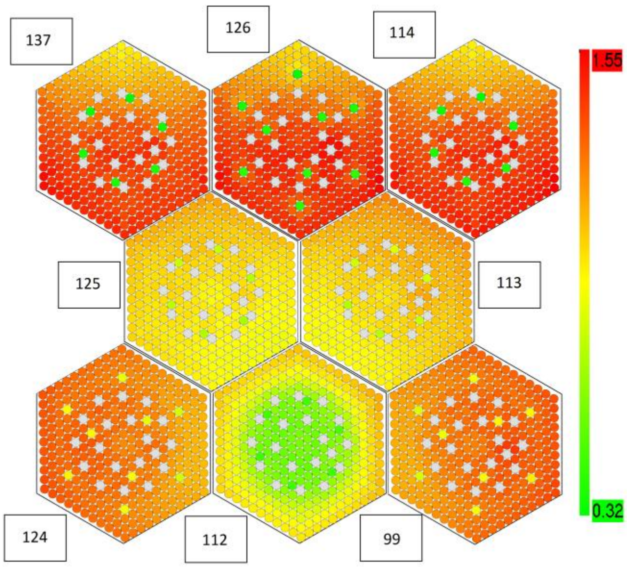

Figure 5. The power distribution (rel. units) in the steady-state (Kr)

Therefore, it is possible that the maximum power value for fuel assembly № 99 is obtained in a fuel rod located on the border with fuel assembly № 112. Then the power value of the fuel rod obtained using a fuel assembly pin-by-pin model is estimated incorrectly.

Now, it is not yet possible to use different feedback parameter values for each fuel rod. Therefore, to assess the power maximum the following actions were implemented. If in the 3D calculation using "hot channel" model for one fuel assembly we reduce artificially the fuel temperature and, accordingly, increase the coolant density to the values of the feedback parameters of the "cold" pin, we get a very conservative estimate of the power for all the pins in this fuel assembly. But in this case, it is necessary to consider not the most heat-loaded fuel rod and not the entire fuel assembly, but one fuel rod, in which the maximum power distribution during accident is observed. However, even in this case, the maximum fuel rod power is less than in the assembly № 114 and is 210 kW. In fuel assembly № 114 power redistribution inside the fuel assembly during the accident does not occur.

It should be noted that the full nodalization of the reactor core with "hot channel" models gives significantly more conservative results. Figure 6 shows the maximum fuel temperature for 99 and 114 fuel assemblies obtained using different models. The calculation results show that the fuel temperature obtained using the pin-by-pin model of fuel assembly is lower than the temperature obtained using the "hot channel" model. Thus, the pin-by-pin model can be used as an additional tool in safety analyses. It allows you to confirm the results, to clarify the calculations parameters and to get more detailed information about fuel rods.

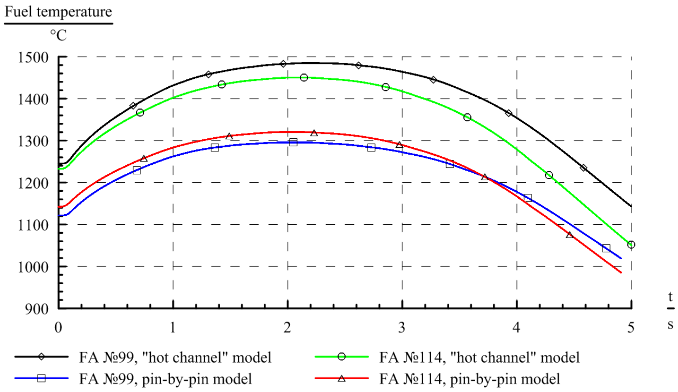

Figure 6. The maximum fuel temperature

CONCLUSION

This work was devoted to the analysis of the FA pin-by-pin model.

The results of simulation of the accident "Control rod ejection" are presented. Calculations have shown that the fuel parameters obtained using the pin-by-pin model are less conservative compared to the parameters obtained using the "hot channel" model. However, the results allow us to conclude that the pin-by-pin model provides more detailed information about the fuel rods in the fuel assemly. In this case, the calculation was conducted with realistic values of the initial parameters, the uncertainties leading to even worse results were not taken into account. Probably, the effect obtained from the pin-by-pin model will be stronger for conservative calculation. In addition, this accident should be considered on zero power because in this case the effect of feedback on the fuel temperature is smaller.

The application of the pin-by-pin model in some accidents is justified, in particular with the control rod ejection, that is an accident of the RIA class (Reactivity-initiated Accident) and requires knowledge about the behavior of the neutron flux in the core. The possibility and necessity of using this model in other accidents requires additional research.

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Rewetting Analysis of a Nuclear Fuel Plate Using an Improved Lumped Model

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I. Introduction

Rewetting is the physical phenomena of a cooling liquid quench front touching a higher temperature surface. Several studies have been performed on the topic in order to help prevent core meltdown in Pressurized Water Reactors (PWRs) during a Loss of Coolant Accident (LOCA).

Experimental [1] and analytical [2, 3, 4, 5, 6] studies have been performed in order to determine the critical heat flux value necessary to have the rewetting phenomena. The effects of precursory cooling [3], surface initial temperature, and Biot number values [5] showed the relation of these variables with the quench front velocity and temperature profiles. The consideration of an internal heat generation source was performed by Sahu et al. [5], using the Heat Balance Integral Method, showing that for low values of the Biot number, the two-dimensional model converged to the uni-dimensional model, and by Pereira and Su [6, 7], showing that the Hermite approximations performed a good approximation for solving the coupled heat transfer equations.

This work develops a model using the Improved Lumped System Analysis (ILSA), by using the Hermite approximations. These approximations consist in interpolating polynomials of the function and its derivatives. It has been shown by previous works [8] that there is an improvement on the results when compared to the Classical Lumped System Analysis (CLSA) model for higher Biot numbers (more than 0.1), given that the physical properties of the boundary are considered in the Hermite approximations. The solutions for the dry and wet temperature distributions were obtained analytically by combination of both solutions at the wet front and the Peclet number was obtained as a function of the Biot number and the dimensionless volumetric heat generation rate.

II. Analysis

A. Mathematical Model

Let us consider the rewetting process of a nuclear fuel plate, initially at temperature T_w, by a coolant at a constant saturation temperature $T_s < T_w$. The flow around the symmetry line is axisymmetric and approximately quasi-steady, the heat generation rate q''', fuel density ρ_f, specific heat c_{pf}, and thermal conductivity k_f are assumed to be constant. After eliminating the temporal dependence by changing the coordinate system to one moving with the rewetting front, the two-dimensional heat conduction equation is rewritten for rectangular coordinates as:

$$\frac{\partial^2 T_f(r,z)}{\partial r^2} + \frac{\partial^2 T_z(r,z)}{\partial z^2} + \frac{\rho_f c_{pf} u}{k_f}\frac{\partial T_f(r,z)}{\partial z} + \frac{q'''}{k_f} = 0, \quad (1)$$

subjected to the following boundary conditions:

$$\left.\frac{\partial T_f(r,z)}{\partial r}\right|_{r=0} = 0, \text{ for } -\infty < z < \infty; \quad (2)$$

$$-k_f \left.\frac{\partial T_f(r,z)}{\partial r}\right|_{r=R_{fo}} = h_1\big(T_f(r,z) - T_s\big), \text{ for } -\infty < z \le 0; \quad (3)$$

$$-k_f \left.\frac{\partial T_f(r,z)}{\partial r}\right|_{r=R_{fo}} = h_2\big(T_f(r,z) - T_w\big), \text{ for } 0 \le z < \infty, \quad (4)$$

where T_f is the fuel temperature, h_1 and h_2 are the heat transfer coefficients for the wet and dry regions, respectively, and R_{fo} is half of the fuel plate thickness.

The following dimensionless variables are introduced:

$$\bar{z} = \frac{z}{R_{fo}}, \bar{r} = \frac{r}{R_{fo}}, Pe = \frac{\rho_f c_{pf} R_{fo} u}{k_f}, Bi_1 = \frac{h_1 R_{fo}}{k_f}, Bi_2 = \frac{h_2 R_{fo}}{k_f},$$
$$\theta_f(\bar{r},\bar{z}) = \frac{T_f(r,z) - T_s}{T_0 - T_s}, \theta_1 = \frac{T_w - T_0}{T_0 - T_s}, Q = \frac{q''' R_{fo}^2}{k_f (T_0 - T_s)}, \quad (5)$$

where T_0 is the wet front temperature corresponds to the temperature achieved by the critical heat flux.

The mathematical model in dimensionless form is given by:

$$\frac{\partial^2 \theta_f(\bar{r},\bar{z})}{\partial \bar{r}^2} + \frac{\partial^2 \theta_f(\bar{r},\bar{z})}{\partial \bar{z}^2} + Pe\frac{\partial \theta_f(\bar{r},\bar{z})}{\partial \bar{z}} + Q = 0; \quad (6)$$

$$\left.\frac{\partial \theta_f(\bar{r},\bar{z})}{\partial \bar{r}}\right|_{\bar{r}=0} = 0, \text{ for } -\infty < \bar{z} < \infty; \quad (7)$$

$$-\left.\frac{\partial \theta_f(\bar{r},\bar{z})}{\partial \bar{r}}\right|_{\bar{r}=1} = Bi_1 \theta_f(\bar{r},\bar{z}), \text{ for } -\infty < \bar{z} \le 0; \quad (8)$$

$$\left.\frac{\partial \theta_f(\bar{r},\bar{z})}{\partial \bar{r}}\right|_{r=R_{fo}} = Bi_2\big(\theta_f(\bar{r},\bar{z}) - (1+\theta_1)\big),$$

for $0 \leq \bar{z} < \infty$. (9)

The average temperature of the fuel plate as a longitudinal position z is defined as follows:

$$\theta_{fav}(\bar{z}) = \int_0^1 \theta_f\,(\bar{r},\bar{z})d\bar{r}. \quad (10)$$

Integrating the governing equation (6) in the transversal direction and applying the definition of the average temperature (10) and the boundary conditions (7) to (9), we have the governing equations for the average temperature $\theta_{fav}(\bar{z})$ respectively in the dry and wet regions:

$$\frac{d^2\theta_{fav}(\bar{z})}{d\bar{z}^2} + Pe\frac{d\theta_{fav}(\bar{z})}{d\bar{z}} + Q - Bi_1\theta_f(1,\bar{z}) = 0,$$

for $-\infty < \bar{z} \leq 0$; (11a)

and

$$\frac{d^2\theta_{fav}(\bar{z})}{d\bar{z}^2} + Pe\frac{d\theta_{fav}(\bar{z})}{d\bar{z}} + Q - Bi_2\left(\theta_f(1,\bar{z}) - (1+\theta_1)\right) = 0,$$

for $0 \leq \bar{z} < +\infty$; (11b)

The surface temperature $\theta_f(1,\bar{z})$ will be expressed as a function of the average temperature $\theta_{fav}(\bar{z})$ by using the lumped system analysis.

The solutions presented on this paper are based on the Classical Lumped System Analysis (CLSA) and Improved Lumped System Analysis (ILSA) with the Hermite approximations. The mathematical model is as follows:

$$\theta_f(0,\bar{z}) = \theta_f(1,\bar{z}) = \theta_{fav}(\bar{z}), \quad (12)$$

for the CLSA and

$$H_{0,0} \rightarrow \int_0^1 \theta_f(\bar{r},\bar{z})dz \approx \frac{1}{2}\left(\theta_f(0,\bar{z}) + \theta_f(1,\bar{z})\right); \quad (13a)$$

$$H_{1,1} \rightarrow \int_0^1 \theta_f(\bar{r},\bar{z})dz \approx \frac{1}{2}\left(\theta_f(0,\bar{z}) + \theta_f(1,\bar{z})\right) +$$

$$+\frac{1}{12}\left(\frac{\partial\theta_f(1,\bar{z})}{\partial\bar{r}} - \frac{\partial\theta_f(0,\bar{z})}{\partial\bar{r}}\right), \quad (13b)$$

for the ILSA.

B. *Classical Lumped Parameters Solution*

By applying the equation (12) into the equations (11a) and (11b), it is found:

$$\theta_{fav}(\bar{z}) = \frac{Q}{Bi_1} + \left(1 - \frac{Q}{Bi_1}\right)e^{\frac{1}{2}\left(-Pe+\sqrt{4Bi_1+Pe^2}\right)\bar{z}},$$

for $-\infty < \bar{z} \leq 0$; (14a)

and

$$\theta_{fav}(\bar{z}) = 1 + \theta_1 + \frac{Q}{Bi_2} + \left(\theta_1 + \frac{Q}{Bi_2}\right)e^{\frac{1}{2}\left(-Pe-\sqrt{4Bi_1+Pe^2}\right)\bar{z}},$$

for $0 \leq \bar{z} < +\infty$. (14b)

C. *Improved Lumped Parameters Solution $H_{0,0}/H_{0,0}$ Formulation*

By applying the equation (13a) into the equations (11a) and (11b), it is found:

$$\theta_{fav}(\bar{z}) = \frac{(4+Bi_1)Q}{4Bi_1} + \left(1 - \frac{(4+Bi_1)Q}{4Bi_1}\right)e^{\frac{1}{2}\left(-Pe+\sqrt{\frac{16Bi_1}{4+Bi_1}+Pe^2}\right)\bar{z}},$$

for $-\infty < \bar{z} \leq 0$; (15a)

and

$$\theta_{fav}(\bar{z}) = 1 + \theta_1 + \frac{Q}{Bi_2} + \frac{Q}{4} +$$

$$+\left(\theta_1 + \frac{Q}{Bi_2} + \frac{Q}{4}\right)e^{\frac{1}{2}\left(-Pe-\sqrt{\frac{16Bi_1}{4+Bi_1}+Pe^2}\right)\bar{z}},$$

for $0 \leq \bar{z} < +\infty$. (15b)

D. *Improved Lumped Parameters Solution $H_{1,1}/H_{0,0}$ Formulation*

By applying the equations (13a) and (13b) into the equations (11a) and (11b), it is found:

$$\theta_{fav}(\bar{z}) = \frac{(3+Bi_1)Q}{3Bi_1} + \left(1 - \frac{(3+Bi_1)Q}{3Bi_1}\right)e^{\frac{1}{2}\left(-Pe+\sqrt{\frac{12Bi_1}{3+Bi_1}+Pe^2}\right)\bar{z}},$$

for $-\infty < \bar{z} \leq 0$; (16a)

and

$$\theta_{fav}(\bar{z}) = 1 + \theta_1 + \frac{Q}{Bi_2} + \frac{Q}{3} + + \left(\theta_1 + \frac{Q}{Bi_2} + \frac{Q}{3}\right)e^{\frac{1}{2}\left(-Pe-\sqrt{\frac{12Bi_1}{3+Bi_1}+Pe^2}\right)\bar{z}},$$

for $0 \leq \bar{z} < +\infty$. (16b)

III. RESULTS

A. *Physical Properties*

Typical PWR operating parameters [9] and fuel physical properties used were: $k_f = 2.163\ W/m.K$, $\rho_f = 10{,}970\ kg/m^3$, $R_{fo} = 4.1\ mm$, $q''' = 318.121\ MW/m^3$. It is assumed that after 100 seconds after a LOCA, the reflooding phase starts and the remaining power inside the core is around 3.5% [9].

B. *Dimensionless Variables*

The dimensionless variables used were defined as the same used by other researchers [5, 6, 7].

TABLE I. DIMENSIONLESS VARIABLES.

Dimensionless Variable	Value
Bi_1	$10.0, 1.0\ and\ 0.1$
Bi_2	$0.001Bi_1$
Q	$1, 10^{-1}, 10^{-2}, 10^{-3}, 10^{-4}\ and\ 10^{-5}$
θ_1	$0.5, 0.9\ and\ 1.5$

C. *Temperature and Rewetting Results*

Figure 1 shows the dimensionless average temperature profile of the fuel plate, using the $H_{1,1}/H_{0,0}$ ILSA. The trends between the CLSA and the ILSA formulations are identical. The gradient temperature near the wet front can be explained by the fact that the wet region temperature increases by a factor of $\frac{Q}{Bi_1}$ and the dry region by a factor of $\frac{Q}{Bi_2}$.

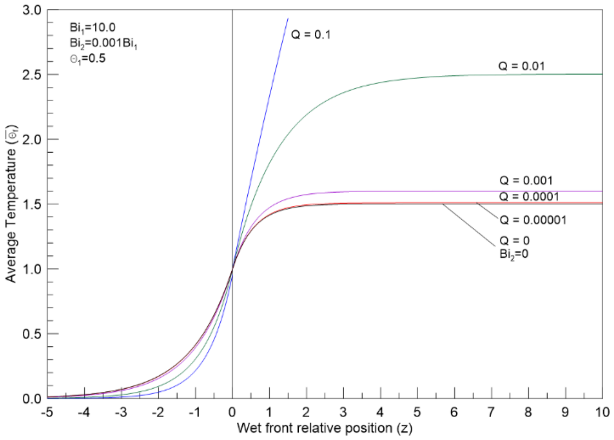

Figure 1. Fuel plate dimensionless average temperature profile variation ($H_{1,1}$ approximation to the fuel temperature and $H_{0,0}$ to the derivative).

Figure 2 shows the comparison between the present work and the solutions found by Sahu et al. [5] using the Heat Balance Integral Method technique (HBIM). As it can be seen, the higher the degree of the ILSA formulation, the closer the solution approaches the HBIM.

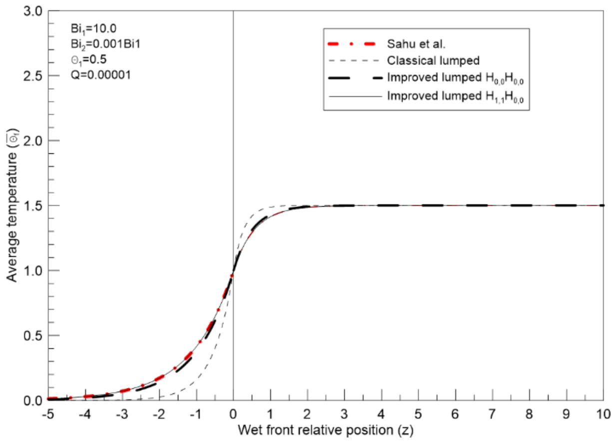

Figure 2. Fuel plate dimensionless average temperature profile comparison of the different formulations with the heat source parameter of 10^{-5}.

When comparing figures 2 and 3, it can be seen that the change of heat source parameter does not appear to influence the relative difference between the solutions.

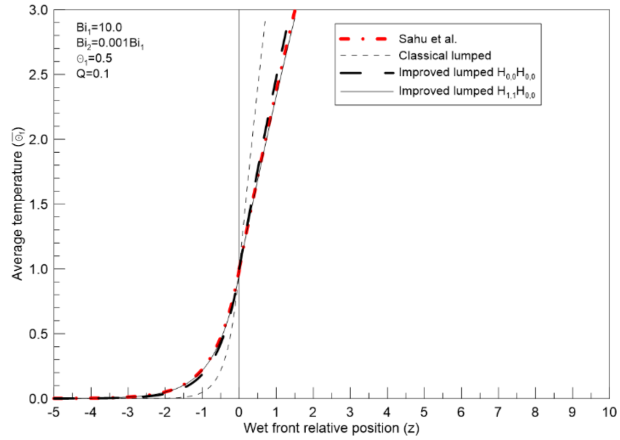

Figure 3. Fuel plate dimensionless average temperature profile comparison of the different formulations with the heat source parameter of 10^{-1}.

Table II shows Peclet values for $Bi_1 = 10.0$.

TABLE II. PECLET NUMBERS WITH $Bi_1 = 10.0$.

Formulation	Heat Source Parameter					
	10^{-5}	***10^{-4}***	***10^{-3}***	***10^{-2}***	***10^{-1}***	***1***
CLSA	3.65	3.60	3.23	1.627	0.252	-
ILSA $H_{0,0}/H_{0,0}$	1.946	1.923	1.720	0.860	0.0852	-
ILSA $H_{1,1}/H_{0,0}$	1.748	1.727	1.545	0.770	0.0615	-
Sahu et al. [5]	1.748	1.727	1.546	0.774	0.0688	-

The increase of heat source parameter decreases the Peclet number due to the increase of the conductive effects compared to the convective effects, hitting a critical value Q_{crit} that stops the rewetting effect.

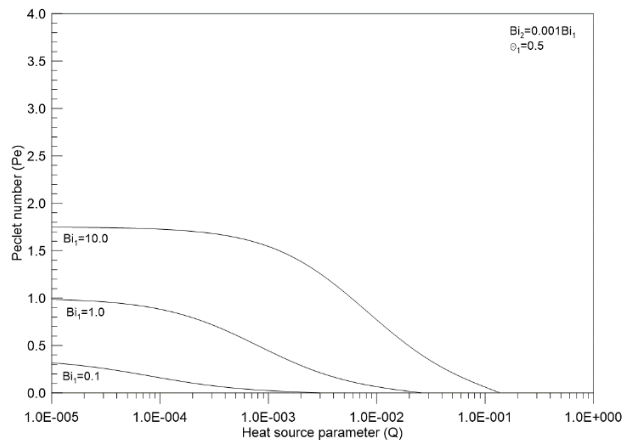

Figure 4. Peclet number variation for diverse heat source parameters and Biot numbers for the ILSA $H_{1,1}/H_{0,0}$ formulation.

The Biot number effect on the rewetting rate for the $H_{1,1}/H_{0,0}$ ILSA can be seen in figure 4 for $\theta_1 = 0.5$. It can be seen that the rewetting rate decreases as the heat source parameter increases, as the higher the Biot number, the greater is the rewetting of the fuel plate. Figure 5 compares different solutions for $Bi_1 = 10.0$ and $\theta_1 = 0.5$.

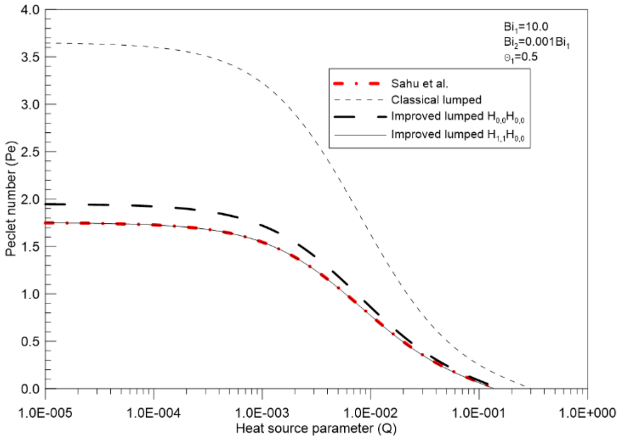

Figure 5. Peclet number variation for diverse heat source parameters and formulations for $Bi_1 = 10.0$ and $\theta_1 = 0.5$.

For $Bi_1 = 0.1$ and $\theta_1 = 0.5$, figure 6 shows that for low Biot, the different formulations converge with reduced differences.

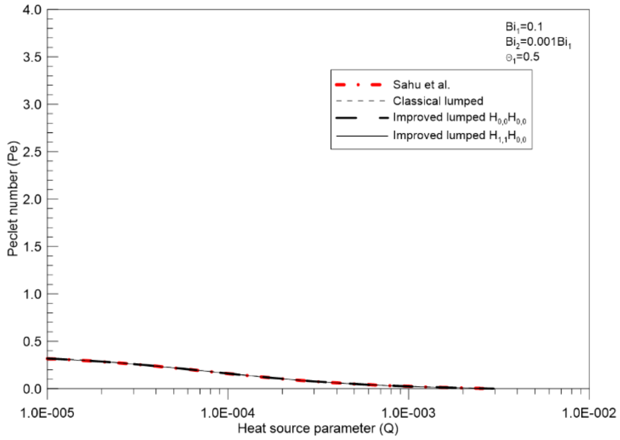

Figure 6. Peclet number variation for diverse heat source parameters and formulations for $Bi_1 = 0.1$ and $\theta_1 = 0.5$.

It can also be seen that the CLSA overestimates the rewetting rate for higher Biot numbers when compared to the $H_{1,1}/H_{0,0}$ formulation, converging to the same results when analyzing lower Biot numbers.

Table III shows the Q_{crit} values to the different formulations.

TABLE III. Q_{crit} VALUES FOR THE CLASSICAL AND IMPROVED LUMPED FORMULATIONS

Formulation	**Biot Number**		
	10.0	***1.0***	***0.1***
Classical lumped	0.295	0.0295	0.00295
Improved lumped $H_{0,0}/H_{0,0}$	0.1514	0.0263	0.00295
Improved lumped $H_{1,1}/H_{0,0}$	0.1349	0.0257	0.00295
Sahu et al. [5]	0.1445	0.0257	0.00295

IV. CONCLUSIONS

The Improved Lumped System Analysis (ILSA) using $H_{0,0}/H_{0,0}$ and $H_{1,1}/H_{0,0}$ approximations for the quasi-stead heat transfer in a fuel plate with internal heat generation showed great convergence with the Classical Lumped System Analysis (CLSA) for low Biot numbers and presumably better results when compared to the Heat Balance Integral Method [5], having lesser computational times. The model described in this paper is important to the design of Light Water Reactors operating with fuel plate by providing predictive design parameters of the core behavior.

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Non-hyperbolicity of RELAP5 two fluid model under loss of flow accident for a boiling water reactor

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I. INTRODUCTION

The IAEA approved nuclear reactor safety analysis code RELAP5 is one of the most widely used code for simulation of the thermal-hydraulic characteristics of light water nuclear reactors under steady-state, transient and accident conditions. The code uses a two-fluid one dimensional, non-equilibrium, non-homogeneous two-phase flow model for thermal-hydraulic simulations. This two fluid model is governed by six phasic conservation equations to define the mass, momentum and energy of the two fluids. It is well-known that two fluid model equation systems may have complex characteristic roots, depending on the fluid conditions which may lead to an ill-posed initial-value problem [1,2,3]. In order to obtain the stable numerical solution the model should be well-posed (hyperbolic having real characteristic roots). The scope of this work consists in finding the non-hyperbolic region of the code model for a boiling water reactor (BWR) under loss of flow accident (LOFA) by the method of characteristics.

The method of characteristics is a simple and fast tool for determining the non-hyperbolicity of a system of differential equations by finding the complex eigenvalues, which has direct implications on obtaining a convergent and stable numerical solution. Ill-posedness is of little concern physically, since the addition of any second-order differential effect (regardless of how small), such as viscosity or surface tension, results in a well-posed problem. The ill-posed (non-hyperbolic) nature is of great concern numerically, since it is necessary that the numerical problem must be well-posed to obtain a stable solution. The approximations inherent in any numerical scheme modify the solution somewhat (truncation error) so these effects can be either stabilizing or destabilizing. The resulting numerical scheme must be stable for mesh sizes of practical interest.

II TWO FLUID MODEL EQUATIONS

The basic conservation equations are not solved directly in RELAP5; instead these equations are solved by transforming them into more convenient set of partial differential equations, also known as "Numerically Convenient Set of Differential Equations". The reason for using this form is ease of degeneration of the model to the single phase case and to increase the elliptic nature of the two fluid model of RELAP5. These equations are available in the research article [4] and a summary of these equations are given here for convenience.

$$\alpha_g \frac{\partial \rho_g}{\partial t} + \alpha_f \frac{\partial \rho_f}{\partial t} + \left(\rho_g - \rho_f\right) \frac{\partial \alpha_g}{\partial t} + \frac{1}{A}\frac{\partial}{\partial x}\left(\alpha_g \rho_g V_g A + \alpha_f \rho_f V_f A\right) = 0 \qquad (1)$$

$$\alpha_g \frac{\partial \rho_g}{\partial t} - \alpha_f \frac{\partial \rho_f}{\partial t} + \left(\rho_g + \rho_f\right) \frac{\partial \alpha_g}{\partial t} + \frac{1}{A}\frac{\partial}{\partial x}\left(\alpha_g \rho_g V_g A - \alpha_f \rho_f V_f A\right) = -\frac{2[H_{ig}\frac{P_s(T^S - T_g)}{P} + (T^S - T_f)H_{if}}{h_g^* - h_f^*} + 2\Gamma_w \qquad (2)$$

Equations (1) and (2) are derived from Vapor and Liquid mass balance equations. In equations (1) and (2), α is the void fraction, ρ is the fluid density, A is the cross-sectional area, V is the velocity of fluid, H_i is interface heat transfer coefficients per unit volume respectively, 'P' is the pressure, P_s is saturation pressure, T_f, T_g and T^s are liquid, vapor and saturation temperatures respectively , h_f^* and h_g^* are liquid and gaseous phase enthalpies associated with bulk interface mass transfer respectively, Γ_W is the vapor mass flux due to wall heat transfer effects, t is the time, x is the spatial distance, and subscripts f and g stand for liquid and vapor phase, respectively.

$$\alpha_g \rho_g \frac{\partial V_g}{\partial t} + \alpha_f \rho_f \frac{\partial V_f}{\partial t} + \frac{1}{2}\alpha_g \rho_g \frac{\partial V_g^2}{\partial x} + \frac{1}{2}\alpha_f \rho_f \frac{\partial V_f^2}{\partial x} = -\frac{\partial P}{\partial x} + \rho_m B_x - \left(\alpha_g \rho_g\right) FWG(V_g) - (\alpha_f \rho_f) FWF(V_f) - \Gamma_g\left(V_g - V_f\right) - HLOSSG * V_g - HLOSSF * V_f \qquad (3)$$

$$\frac{\partial V_g}{\partial t} - \frac{\partial V_f}{\partial t} + \frac{1}{2}\frac{\partial {V_g}^2}{\partial x} - \frac{1}{2}\frac{\partial {V_f}^2}{\partial x} = -\left(\frac{1}{\rho_g} - \frac{1}{\rho_f}\right)\frac{\partial P}{\partial x} - FWG * V_g + FWF * V_f + \frac{\Gamma_g[\rho V_I - (\alpha_g \rho_g V_f + \alpha_f \rho_f V_g)]}{\alpha_g \rho_g \alpha_f \rho_f} - \rho_m FI\left(V_g - V_f\right) - \frac{C\rho_m^2}{\rho_g \rho_f}\frac{\partial}{\partial t}\left(V_g - V_f\right) + \frac{\rho_m}{\rho_g \rho_f}\left(\rho_f - \rho_g\right) B_y \frac{dy}{dx} - HLOSSG * V_g + HLOSSF * V_f \qquad (4)$$

Equations (3) and (4) are derived from Vapor and Liquid momentum balance equations. In equations (3) and (4), ρ_m is the density of mixture, B_x is the body force in X direction, V_I is the interfacial velocity, FWF and FWG are wall drag coefficients for liquid and vapor phase respectively, Γ is volumetric mass exchange rate, Γ_g is vapor generation rate, Γ_{ig} is the mass transfer at the vapor/liquid interface in the bulk fluid, FI is inter-phase drag coefficient, C is the virtual mass coefficient and other symbols used are defined previously.

$$\left(\rho_g u_g + P\right)\frac{\partial\alpha_g}{\partial t} + \alpha_g u_g \frac{\partial\rho_g}{\partial t} + \alpha_g\rho_g\frac{\partial u_g}{\partial t} + \frac{1}{A}\left[\frac{\partial}{\partial x}\left(\alpha_g\rho_g u_g V_g A\right) + P\frac{\partial}{\partial x}\left(\alpha_g V_g A\right)\right] = -\left(\frac{h_f^*}{h_g^*-h_f^*}\right)\frac{P_s}{P}H_{ig}\left(T^s - T_g\right) - \left(\frac{h_g^*}{h_g^*-h_f^*}\right)H_{if}\left(T^s-T_f\right) - \left(\frac{P-P_s}{P}\right)H_{gf}\left(T_g-T_f\right) + \left[\left(\frac{1+\epsilon}{2}\right)h_g' + \left(\frac{1-\epsilon}{2}\right)h_f'\right]\Gamma_W + Q_{Wg} + DISS_g \qquad (5)$$

$$-\left(\rho_f u_f + P\right)\frac{\partial\alpha_g}{\partial t} + \alpha_f u_f \frac{\partial\rho_f}{\partial t} + \alpha_f\rho_f\frac{\partial u_f}{\partial t} + \frac{1}{A}\left[\frac{\partial}{\partial x}\left(\alpha_f\rho_f u_f V_f A\right) + P\frac{\partial}{\partial x}\left(\alpha_f V_f A\right)\right] = \left(\frac{h_f^*}{h_g^*-h_f^*}\right)\frac{P_s}{P}H_{ig}\left(T^s - T_g\right) + \left(\frac{h_g^*}{h_g^*-h_f^*}\right)H_{if}\left(T^s-T_f\right) + \left(\frac{P-P_s}{P}\right)H_{gf}\left(T_g-T_f\right) - \left[\left(\frac{1+\epsilon}{2}\right)h_g' + \left(\frac{1-\epsilon}{2}\right)h_f'\right]\Gamma_W + Q_{Wf} + DISS_f \qquad (6)$$

Equations (5) and (6) are derived from Vapor and Liquid energy balance equations. In equations (5) and (6), u_g and u_f are specific internal energies of gaseous and liquid phase respectively, H_{gf} is sensible (direct) heat transfer coefficient per unit volume, h_g' and h_f' are phasic specific enthalpies of vapor and liquid associated with wall (thermal boundary layer) interface mass transfer, respectively, ϵ is surface roughness, Q_{Wg} and Q_{Wf} are gaseous and liquid phase wall heat transfer rates per unit volume, respectively, $DISS_g$ and $DISS_f$ are energy dissipation functions of gaseous and liquid phase respectively and other symbols used are defined previously.

III RESULTS AND ANALYSIS

Figure 1 shows the thermal-hydraulic nodalization of a standard Laguna Verde BWR reactor. This input model includes a reactor vessel, recirculation loops, and main steam supply system. Steam separator and steam dryer assemblies are modeled as equivalent individual components. Two recirculation loops have been modeled using generic RELAP components for recirculation pumps, valves, and jet pumps. A single jet pump is modeled as equivalent to 10 Laguna Verde jet pumps in both recirculation loops. Physical and thermodynamic properties of both the loops are identical. The input data used here is the same as what was used in a previous study for station black-out [5].

Under the postulated loss of flow accident scenario we have, assumed that the mass flow rate of cooling loop is reduced to very low. The reactor is shut down by control rods but the heat is still adding to the fluid due to available decay heat. The pressure and temperature of coolant increasing continuously and results in increasing void fraction in the core. Since there is no force flow, the liquid water velocity in core channels will be very low but due to the buoyancy force the steam still goes upward with some velocity, resulting in large slip ratio.

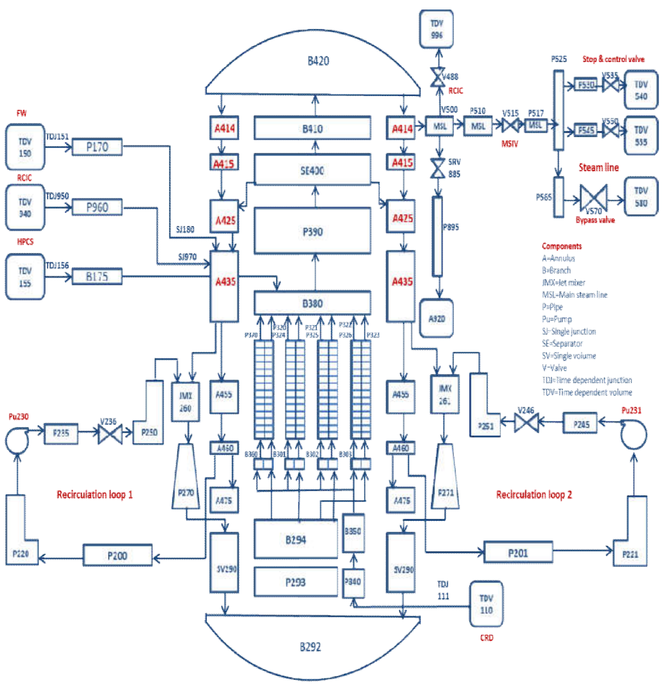

Figure 1. Thermal-hydraulic nodalization of a standard Laguna Verde BWR reactor

We have applied method of characteristics to the RELAP5 set of equations. The steady state initial conditions of the system are (Pressure 7 MPa and Temperature 298°C). Fig. 2, 3 and 4 shows the region of complex characteristics (non-hyperbolic region) for different sample fluid conditions under loss of flow accident (LOFA). Fig. 2 shows the region of complex characteristics for (V_f=2.25 m/s, V_g varies 0.25 to 2.5 m/s, when pressure rises to 7.5 MPa and fluid temperature rises to 305°C), Fig. 3 shows the region of complex characteristics for (V_f=0.1 m/s, V_g varies 0.1 to 10 m/s, when pressure rises to 8 MPa and fluid temperature rises to 308°C) and Fig. 4 shows the region of complex characteristics for (V_f=0.01 m/s, V_g varies 0.01 to 10 m/s, when pressure rises to 8.5 MPa and fluid temperature rises to 310°C). As shown in Fig.2,3 and 4, for the slip ratio about unity, the characteristics of system is real for all values of void fraction hence the system of equations is hyperbolic and produce stable numerical solution thus produce accurate results (this is the case for steady-state operation which is defined by close to unity slip ratio with large mass flow rate). It has been also seen for larger value of slip ratio and void fraction the characteristics of the system is complex and hence the numerical solution thus obtain is inaccurate and having oscillations due to ill-posed (non-hyperbolic) model (this is the case for LOFA operation which is defined by large slip ratio with very low mass flow rate and liquid phase velocity V_f). Here we have only taken the case of constant liquid velocity and varied the vapour velocity for each case of fig. 2 to 4, since we have observed that the region of complex characteristics is only varies with relative velocity (V_g-V_f). It is common observation under these sample fluid conditions that the oscillations occurs in RELAP5 simulation results. This can be explained complex characteristics of system for large slip ratios as shown in Fig. 2 to 4. Another observation is the sudden termination of RELAP5 simulation with a message "thermal hydraulic property error", and having comparatively large mass error and long execution time in RELAP simulation results of BWR in case of such sample fluid conditions. These issues can also be explained by complex characteristics of the system, as in LOFA scenario void fraction and slip ratio increase to large values.

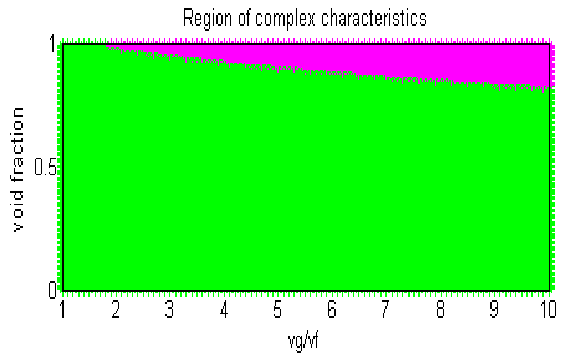

Figure 2. Region of complex characteristics for sample fluid conditions(V_f=2.25, V_g=2.25 to 22.5) "*" complex "*" real

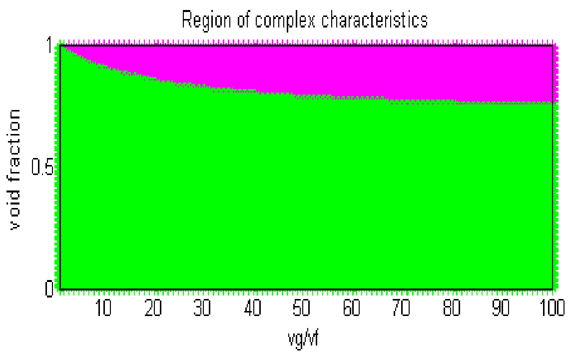

Figure 3. Region of complex characteristics for sample fluid conditions (V_f=0.1, V_g=0.1 to 10)) "*" complex "*" real

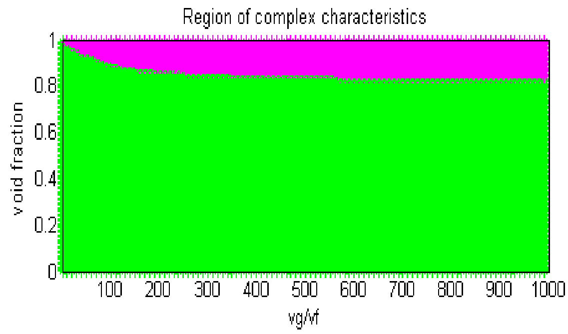

Figure 4. Region of complex characteristics for sample fluid conditions (V_f=0.01, V_g=0.01 to 10) "*" complex "*" real

The real situation of RELAP5 system of equations is analyzed for a BWR LOFA sample conditions in Fig.5. Fig.5 gives information for variation of real and imaginary part of all 6 roots with the variation of void fraction and slip ratio. There are 6 eigenvalues for each value of void fraction and slip ratio at a constant pressure of 7 MPa. There is a uniform increment in void fraction with step size 0.1 and uniform increment in slip ratio with step increment 1.0, so there are total 600 characteristics roots for Fig.5 or we can say there are 60 roots for each value of slip ratio.

It is observed from the Fig.5 that for smaller value of slip ratios, roots are real in nature (amplitude close to zero) and for larger value of slip ratios roots become complex as seen in Fig.5. It is also observed that amplitude of real and imaginary parts of complex roots and number of complex roots increases (large region of complex characteristics as seen in figure 2,3 and 4) with increase in slip ratio. The effect of these complex roots can be seen in RELAP simulation results.

Void fraction also affects the eigenvalues. There are three cases according to range of void fraction for BWR cases.

(a) for void fraction range ($0 \leq \alpha < 0.5$): all the roots are real and amplitude is very small (~0). The system in this case is hyperbolic and results in unique, accurate and stable solution. (no sensitivity and oscillations issues). Linear stability in literature [1] of the two equation model for bubbly flow including the virtual mass and interfacial pressure forces also confirms that it is well-posed for low void fractions.

(b) for void fraction ($0.5 \leq \alpha \leq 0.75$): Increasing void fraction from 0.5 to 0.75 causes increment in real and imaginary parts of complex roots, while amplitude of real roots is not affected (~0). For void fraction 0.75 imaginary part of complex eigenvalues is maximum. So model with complex roots having void fraction 0.5 and its close neighbor values is highly sensitive to disturbance with high amplitude and frequency oscillations.

(c) for void fraction range ($0.75 < \alpha \leq 1.0$): increase in value of void fraction leads to decrease amplitude of imaginary part while the real part of complex roots keeps increasing with void fraction. There is no effect on amplitude of real roots. For higher value of void fractions along with the increase in slip ratio cause decrease in sensitivity of the system for initial values of parameters but oscillation frequency of disturbances increases in this range.

It has been seen that all the characteristics roots remains real for steady state and initial accident conditions until the slip ratio below 2 and void fraction below 0.5. When the slip ratio is above 2 and void fraction above 0.5 the characteristics roots become complex in nature and it has been seen that out of six characteristics roots only two roots become complex for some thermal hydraulic conditions and remaining 4 roots are always real. These numbers are different for different systems (BWR, AHWR and fusion systems [6,7, 8 and 9] etc.) but the pattern of variation of roots with parameters is always same.

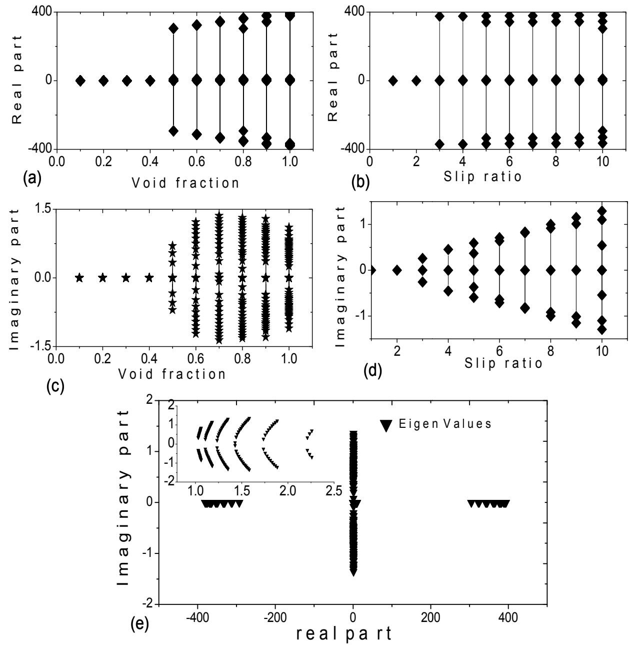

Figure 5. Eigenvalue spectrum for BWR LOFA condition (for velocity (v=1 to 10 m/s), (a) variation of real part of eigenvalues with void fraction, (b) variation of imaginary part of eigenvalues with void fraction, (c) variation of real part of eigenvalues with slip ratio, (d) variation of imaginary part of eigenvalues with slip ratio, (e) Eigenvalues in complex plane.

CONCLUSIONS

We have been able to determined the range of flow regimes and sample fluid conditions where the two fluid model of the code becomes non-hyperbolic for the RELAP5 specific models. The analysis shows that the model become non-hyperbolic for some of the operating spectrum of BWR loss of flow accident. The exact implications of this may not be certain but it provides an insight into the mathematical properties and stability of the model. It has been found that when the void fraction and relative velocity between the liquid and gas exceeds a critical value, the governing equations becomes non-hyperbolic and the model become ill-posed. This ill-posedness suggests that the results of the standard two-fluid model do not reflect the real flow physics inside the channel for those conditions and have numerical issues. The basic two-fluid model of RELAP5 only gives meaningful results when the void fraction and relative velocity between the vapor phase and the liquid phase are less than a critical value, which varies with system pressure, temperature, channel diameter, gravity, and various other flow properties. It is suggested that, to make the two fluid model of RELAP5 unconditionally well-posed the missing physics in conservation equations (like diffusion term and bubble collision force term etc.) should be added and more accurate closure relations are needed.

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CFD Simulations of the OPAL Cold Neutron Source

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I. Introduction

Recent years have seen advances in both computer simulation technology and thermal-hydraulic simulation techniques. Increases in the availability of computing resources such as RAM and parallel processors have made modelling of large and complex systems faster, more affordable and more cost-effective, while recent advances in Computational Fluid Dynamics (CFD) techniques are producing more accurate and reliable simulation results of said complex systems. In particular, the invention of polyhedral meshing techniques has yielded a significant leap in fast and accurate CFD simulations.

Within ANSTO, the Open Pool Australian Light-water (OPAL) reactor's Cold Neutron Source (CNS) provides an ideal application for the latest CFD simulation techniques. Due to its position directly adjacent to the OPAL reactor core, the CNS In-Pile has no in-situ instrumentation that can withstand the extreme environment. Instead, CFD calculations have been performed to characterise the behaviour of the cryogenic helium and liquid deuterium (LD2) within the CNS thermosiphon, with results being compared to prototype experiments performed prior to its installation.

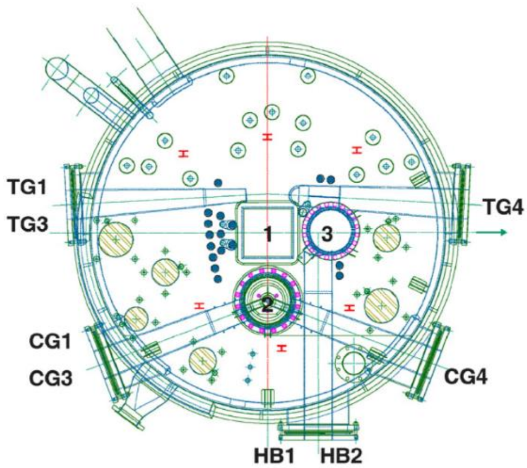

Figure 1: A top-down view schematic of the OPAL reactor pool, showing (1) The reactor core, (2) The Cold Neutron Source, and (3) The Hot Neutron Source, as well as labelled neutron guides.

Previous steady-state CFD simulations of the CNS have required up to three weeks to converge. However, encompassing the latest CFD simulation techniques with the same computing resources has since reduced convergence time to a matter of hours while producing more accurate results.

OPAL's CNS is due to for replacement in 2024, and this new simulation technique provides not only a better and faster method for modelling the performance of the current CNS, but also a useful tool for predicting the performance of proposed future CNS designs.

II. The OPAL Cold Neutron Source

The CNS on the OPAL reactor is a flagship research tool that provides cold neutrons to a number of neutron scattering instruments in the Australian Centre for Neutron Scattering (ACNS), which serves both national and international researchers. A wide range of research fields ranging from biological science to pharmaceutical research to food technology have seen a strong growth in the application of cold neutron scattering in recent years. By international comparison, the OPAL CNS has an extremely high performance and the future of neutron scattering at ANSTO depends critically on the CNS.

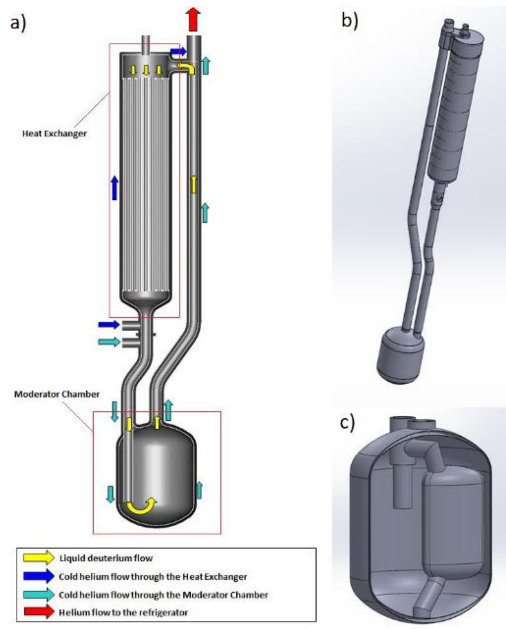

Figure 2: a) A circulation diagram illustrating the flow circuits of helium and deuterium within the OPAL CNS thermosiphon, b) A full CAD rendering of the complete thermosiphon that was used for mesh generation, and c) A cutaway rendering of the CNS moderator chamber, including the helium displacer.

The CNS, located in the heavy water reflector vessel, reduces the energy of the thermal neutrons by moderating them with liquid deuterium at cryogenic temperatures, thus making them 'cold'. Neutron guides adjacent to the CNS transport cold neutrons from the CNS to neutron scattering instruments in the ACNS Neutron Guide Hall.

The CNS's liquid deuterium is maintained in a single phase in what is called the CNS In-Pile, which is a thermosiphon housed within a vacuum containment 'thimble'. The thermosiphon is comprised of an upper stainless steel tube-in-shell heat exchanger and a lower aluminium alloy double-walled moderator chamber. Bi-metal junctions connect the two parts into a single, vertical structure. Under normal operating conditions at full reactor power (20 MW thermal), liquid helium at 20 K is circulated through the thermosiphon to maintain deuterium in its liquid phase while removing 5 kW of heat.

The current CNS was installed in 2004 and has been in operation since the reactor was first commissioned. It is due for replacement by 2024, which will also serve as an opportunity to improve on the existing design. To that end, it has been proposed to increase the height of the CNS moderator chamber by 70 mm, increasing the volume of LD2 by approximately 5 L and thereby maximising the production of cold neutrons.

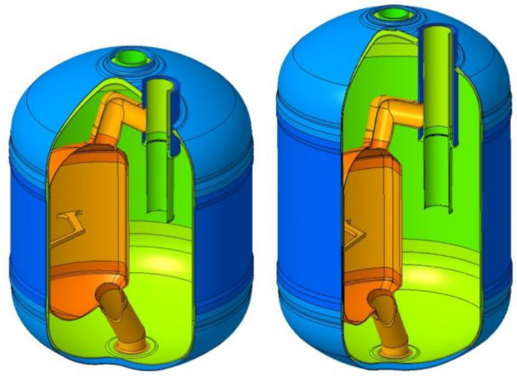

Figure 3: Left – the moderator chamber for the existing OPAL CNS, and Right – the current proposed moderator chamber design, with height increased by 70 mm.

III. Advances in CFD Techniques

As an engineering tool, CFD techniques distretise different continua (e.g. solids, fluids) into individual volume elements that are connected to one another by adjacent faces. This process is known as creating a volume 'mesh'. By applying physical models and their associated equations to each volume cell and its interfaces with other cells in the mesh, simulations can model the behaviour of the fluid (or solid) as a whole. With high-speed computers, better solutions can be achieved, and they are often required to solve the largest and most complex problems.

Numerous commercial products are available for performing CFD simulations. This investigation employs STAR-CCM+ simulation software, which is noted for its incorporation of a 'polyhedral' meshing method. Historically, automated mesh generation software generates either tetrahedral or hexahedral cells, as these are the simplest three-dimensional structures. STAR-CCM+ improves upon this technique by first generating a tetrahedral mesh, then combining tetrahedral cells into larger polyhedral cells. Polyhedral meshing is a genuine advance in automated meshing as it has been shown to be less numerically diffusive and more computationally efficient, with fewer total cells required to capture similar levels of detail.

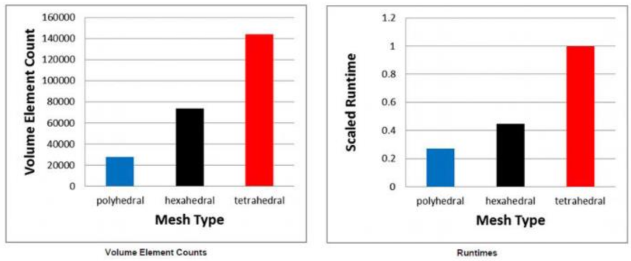

Figure 4: A comparison of polyhedral, hexahedral and tetrahedral meshes for identical volumes shows that polyhedral meshes require less elements and shorter runtimes for simulation convergence.

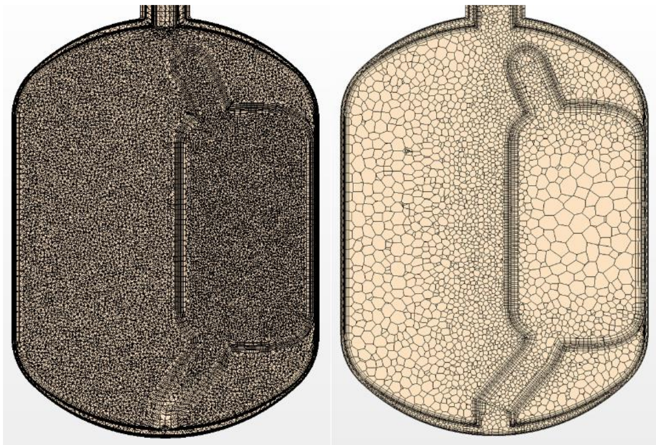

Figure 5: Left – A cross-section of a tetrahedral mesh of the CNS moderator chamber, and Right – A cross-section of the polyhedral mesh of the CNS moderator chamber. The polyhedral mesh requires far fewer cells.

Comparative simulations have shown that, compared to tetrahedral meshes, typically one fifth of the number of polyhedral cells are required to achieve the same accuracy in results. Additionally, simulations performed with polyhedral meshes took approximately half the simulation time required for convergence. Previous simulations of the OPAL CNS, using traditional tetrahedral meshing techniques, have required approximately 20 million cells and one week of simulation time to reach steady state convergence. The application of STAR-CCM+ provides the potential to reduce the number of requires cells to approximately 5 million, while achieving steady state convergence in as low as 10 hours of simulation time.

IV. CNS Prototype Experiments

Prior to the original CNS installation, PNPI prototype experiments were performed on a CNS prototype in 2003, with the goal being to study the behaviour of the prototype thermosiphon at different heat loads. The prototype was a one-to-one scale mock-up of the real OPAL CNS, with the addition of two electrical heaters that were used to simulate the heat load produced by reactor radiation.

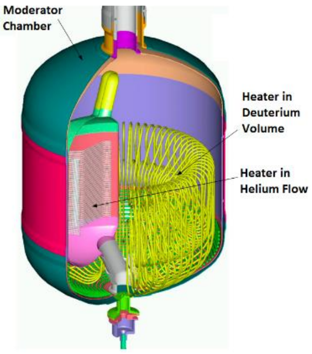

Figure 6: The prototype CNS experiments employed heating coils within the helium and deuterium volumes of the moderator chamber to simulate the expected radiation heat load.

Key measurements that provided a reference for simulation results to be compared included the temperature of the deuterium at the heat exchanger inlet (DT1), the temperature of the deuterium at the heat exchanger outlet (DT2), the temperature of the deuterium at the moderator chamber outlet (DT3) and the temperature of the helium at the thermosiphon outlet (HeT1).

V. Previous Simulation Attempts

First simulations for the steady state operation of the CNS were performed with CFX simulation software and employed HyperMesh software to produce a traditional tetrahedral mesh. Due to memory constraints, a simplified CNS geometry of approximately 11 million tetrahedral cells was created. Helium and deuterium heat loads, helium inlet temperatures and flow rate input variables were all matched to values recorded from the PNPI prototype experiments. Using a 64-parallel-processor cluster, a transient simulation required 8000 iterations and approximately 500 hours to converge to a steady solution.

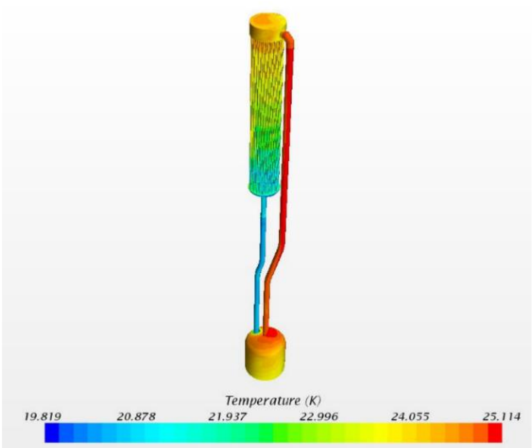

Figure 7: Simulation results from first STAR-CCM+ simulations showing the deuterium volume temperatures.

The same geometry and mesh was later imported into STAR-CCM+ software, and another steady state simulation with identical inputs was performed using only 12 parallel processors. Using the STAR-CCM+ solvers, the simulation required 30,000 iterations and 228 hours to converge, and showed a significant reduction in error in deuterium flow rate and deuterium and helium temperatures within the CNS.

VI. Simulations With STAR-CCM+ Polyhedral Mesh

A more detailed geometry of the complete CNS In-Pile was developed with CAD and imported into the STAR-CMM+ software. This model was used to generate a detailed, native polyhedral mesh comprised of approximately 5 million polyhedral cells. Identical inputs to the previous simulations and PNPI prototype experiments were applied, and the simulation was run on a 12-parallel-processor desktop computer until it achieved steady state convergence. The simulation required only 3500 iterations and 12 hours to converge. These results demonstrate a tenfold increase in efficiency when compared with previous simulations.

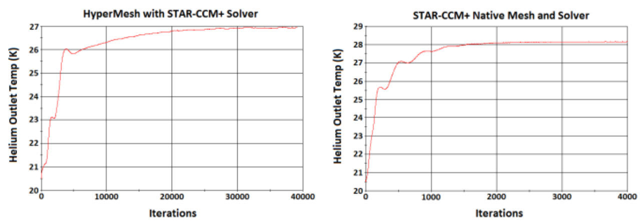

Figure 8: Simulations using the native STAR-CCM+ polyhedral mesh and solver showed a tenfold reduction in simulation runtime.

In addition to the improvements in efficiency, using the STAR-CCM+ native mesh and solver also reduced error in deuterium and helium temperature measurements to below 1% when compared with the PNPI prototype experiments.

TABLE 1: Important simulation and experiment results for the PNPI CNS prototype experiments (PNPI), CFX simulations with tetrahedral mesh (CFX), STAR-CCM+ simulations with external tetrahedral mesh (STAR-T) and STAR-CCM+ simulations with native polyhedral mesh (STAR-P). Simulation results were determined by four key measurements: the temperature of deuterium at the HX inlet (DT1), the temperature of the deuterium at the HX outlet (DT2), the temperature of the deuterium at the MC outlet (DT3), and the temperature of the helium at the thermosiphon outlet.

Variable	PNPI	CFX	STAR-T	STAR-P
DT1	25.10	26.00	24.87	25.25
DT2	22.00	22.64	22.18	22.08
DT3	25.20	26.23	24.87	25.30
HeT1	24.10	23.98	23.50	24.40

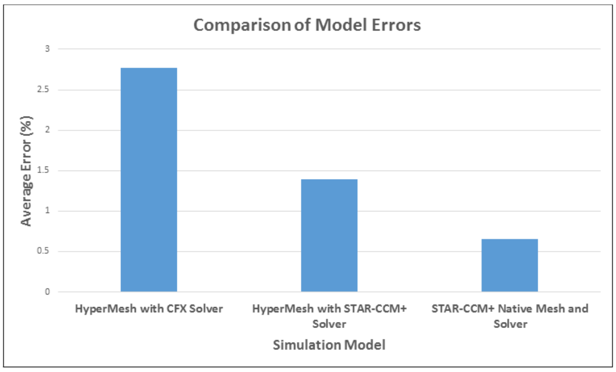

Figure 9: When compared against the PNPI prototype experiments, the STAR-CCM+ native polyhedral mesh and solver produced the lowest error.

Based on neutronics models and reactor operating measurements, simulations were then performed to predict the behaviour of the CNS during normal reactor operation. Given the simulation model's accuracy at reproducing the prototype experiment results, the simulation was used to accurately predict the state of deuterium and helium in the CNS In-Pile under real operating conditions.

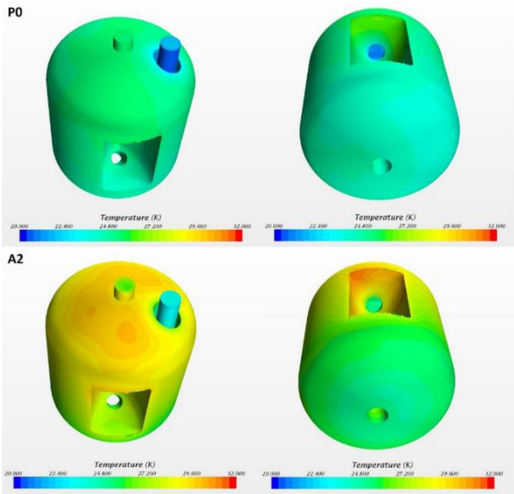

Figure 10: Simulations using known reactor conditions (A2) showed that the deuterium within the CNS moderator chamber is at a higher temperature than that predicted by the PNPI prototype experiments (P0).

VII. Simulations of Future CNS Designs

Given the STAR-CCM+ simulation model's success at reproducing the behaviour of the CNS in operation, as well as its ease and efficiency as an engineering tool, the same method was used to predict the performance of proposed CNS designs under consideration for its replacement in 2024. These simulations are now considered integral when assessing the validity of future CNS performance and proposed design modifications.

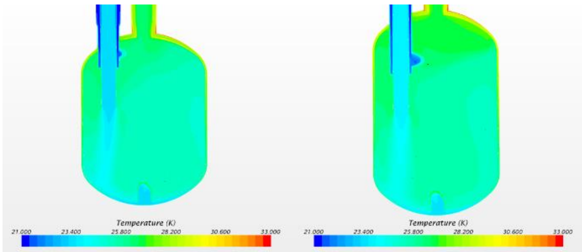

Figure 11: Comparison simulations between the existing MC design (left) and a proposed design with increased MC height (right).

VIII. Conclusions

This paper details the advantages of the latest polyhedral meshing techniques in CFD simulations. STAR-CCM+ software was employed to simulate the normal operating conditions of the OPAL reactor's CNS In-Pile, with simulation results being compared to experimental results of CNS prototype measurements. Results confirmed that polyhedral meshing techniques reduce the memory requirements of simulation computing hardware as well as reducing the time required for comparative steady state simulations to converge. When judged against previous simulation attempts, results from the STAR-CCM+ simulations produced a lower error in measurements of the CNS deuterium and helium temperatures when compared to the PNPI CNS prototype test results.

Due to its high accuracy and efficient simulation techniques, CFD simulations of the OPAL CNS using STAR-CCM+ software provide a relatively cheap tool for predicting the performance of the CNS under different operating conditions, as well as serving as a useful design tool in predicting the performance of proposed future CNS designs.

Acknowledgments

Thanks to Weijian Lu for his help in recreating the CNS structure and operating conditions in simulation and thanks to Mark Ho and Guan Heng Yeoh for their extensive help and expertise on CFD simulation techniques. Credit also goes to Tom Pavlou, Haneol Park and Yeongshin Jeong for their hard work and research in CFD simulations of the OPAL CNS that preceded and enabled this study.

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TRACK 5: NUCLEAR MATERIALS

NUMERICAL ANALYSIS OF EARLY-AGE MASS CONCRETE

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METALLIC FAST REACTOR SAFETY STUDIES

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ENGINEERING MX PRECIPITATION IN FERRITIC/MARTENSITIC GRADE 91 STEEL WITH WIRE ARC ADDITIVE MANUFACTURING

T.M. KELSY GREEN
UNIVERSITY OF MICHIGAN, USA

UTILIZATION OF MINIATURIZED TESTING SPECIMENS FOR THE IRRADIATED REACTOR PRESSURE VESSEL MATERIALS DEGRADATION EVALUATION

K. RUSŇÁKOVÁ, O. BURŠÍK, I. ELIÁŠOVÁ, M. AUGULIS
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DEVELOPMENT OF ATTILHA (ADVANCED TEMPERATURE AND THERMODYNAMICS INVESTIGATION BY A LASER HEATING APPROACH) SET UP FOR HIGH TEMPERATURE APPLICATIONS

A. QUAINI, S. GOSSE, T. ALPETTAZ, C. BONNET, J.-M. BORGARD
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Numerical analysis of early-age mass concrete

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I. Introduction

Nuclear Power Plants (NPP) and other nuclear safety related facilities involve complex civil engineering structures that are essential to their safe and reliable operation. The function of these structures is dual: they house all systems and components providing them an adequate operating environment; they protect the nuclear facilities from external hazards and mitigate the impact of an internal accident.

The most common material used for nuclear safety related structures is reinforced concrete because of its numerous advantageous properties. Amongst these are good shielding abilities against radiation, high mechanical strength, ease of implementation and production and relatively low cost.

Due to the increase of the structures lifetime in, for example, new build NPP, intermediate or final nuclear waste disposal facilities or long-term operation of existing plants, concerns about ageing issues of reinforced concrete have arisen in recent years. Issues such as carbonation, chlorides attack, alkali–silica reactions and others can lead to the loss of required strength needed for the safe operation of the facility.

A key aspect for a sustainable concrete in very thick structures, so-called mass concrete, is the control of the heat development during the highly exothermic reaction of cement hydration at early age. Indeed, excessive temperatures can lead to different concrete pathologies such as delayed ettringite formation (DEF) and cracking which can both lead to a reduced longevity of the structure. As most of the concrete elements in nuclear safety related structures are very thick for shielding purposes and protection against external hazards, the control of heat development is crucial in nuclear civil engineering.

While issues related to excessive heat development in mass concrete are known to most civil engineers, the heat development in mass concrete itself is still often not controlled mainly due to high number of factors influencing this phenomenon and the lack of analysis tools and methods in design codes and software.

The aim of this paper is to present an effective and practical methodology to analyze heat development in mass concrete at early age, making it suitable for a production environment. This methodology can indeed allow the civil engineer to have a better understanding of the numerous parameters which can affect heat development (temperature field) and thermal cracking (tensile stress to tensile strength ratio). The engineer can use it to implement on site mitigation measures such as low heat cements, casting sequence and insulation. The methodology can easily be implemented in a general finite element package capable of both thermal and mechanical analyses using user-defined material models and loadings.

II. Mathematical models

The behavior of early-age concrete is governed by complex physicochemical phenomena which include cement hydration, drying, shrinkage and creep. In the present paper, the behavior is simplified by uncoupling the problem in two distinct parts: first, the cement hydration process is simulated using a thermal formulation. Secondly, the concrete strains are computed using the thermal fields issued from the first part.

A. Cement hydration

The rate of hydration of cement can be described by the general form of solid-state reaction as [1]:

$$\dot{\xi}(t) = A(\xi)\exp\left[-\frac{E_a/R}{T(t)}\right] \qquad (1)$$

Where $A(\xi)$ is the reaction affinity $[s^{-1}]$ depending on the degree of hydration ξ ; E_a is the energy of activation of cement $[\mathrm{J\,mol^{-1}}]$; R is the gas constant $[\mathrm{J\,mol^{-1}K^{-1}}]$; T is the temperature [K] at a time t [s].

The degree of hydration $\xi(t)$ can be expressed by:

$$\xi(t) = \frac{Q(t)}{Q_\infty} \qquad (2)$$

Where $Q(t)$ is the hydration heat $[\mathrm{J/g_{cement}}]$ at time t; Q_∞ is the hydration heat at the end of hydration process.

The reaction affinity in respect of hydration degree, $A(\xi)$, can be deduced from adiabatic tests on concrete samples.

Using (1) and (2), the heat generation rate q $[\mathrm{W\,m^{-3}}]$ of the hydration process can be expressed as:

$$q(t) = CQ_\infty\dot{\xi}(t) = CQ_\infty A(\xi)\exp\left[-\frac{E_a/R}{T(t)}\right] \qquad (3)$$

Where C is the cement content in the concrete $[\mathrm{g_{cement}m^{-3}}]$ and $\xi(t)$ and $T(t)$ can be determined through a time-stepping procedure using a transient thermal finite element analysis.

B. Concrete strain

Total strain $\epsilon(t)$ in concrete can be expressed at any time t as follows:

$$\epsilon(t) = \epsilon_i(t) + \epsilon_c(t) + \epsilon_s(t) + \epsilon_T(t) \qquad (4)$$

Where $\epsilon_i(t)$ is the instantaneous strain at time t; ϵ_c is the creep strain at time t; ϵ_s is the shrinkage strain; ϵ_T is the thermal strain. The two first terms on the right side of equation (4) are stress-dependent strains, the two last ones are stress-independent strains.

Equation (4) is often expressed in incremental form in mechanical finite-element analyses in order to take into account material non-linearity and age-dependent properties:

$$\Delta\epsilon(t) = \Delta\epsilon_i + \Delta\epsilon_c + \Delta\epsilon_s + \Delta\epsilon_T \qquad \text{(4bis)}$$

1) Instantaneous elastic strain

For stresses lower than 40% of the concrete characteristic compressive strength, concrete behaves basically as a linear elastic material. The initial strain increment can thus be determined as an elastic strain increment using Hooke's law:

$$\Delta\epsilon_i(t) = \frac{1}{E_c(t)}[D]\,\Delta\sigma(t) \qquad (5)$$

Where E_c is the time dependent elasticity modulus [N m^{-2}]; $[D]$ is the compliance tensor; $\Delta\sigma$ is the stress increment [N m^{-2}]. Explicit concrete cracking behavior is not considered in this paper for simplification purposes.

2) Creep strain

For stresses lower than 40% of the concrete characteristic compressive strength, concrete behaves as an aging linear visco-elastic material:

$$\Delta\epsilon_c(t) = \Delta\sigma(t_0)\,J(t,t_0) + \int_{t_0}^{t} J(t,\tau)\frac{\partial\sigma(\tau)}{\partial\tau}d\tau \qquad (6)$$

Where $J(t,t_0)$ is the creep compliance function [N^{-1}m^2]; $\Delta\sigma(t_0)$ is the stress increment at the start of time step t_0.

3) Thermal strain

The thermal strain increment depends linearly on the coefficient of thermal expansion and the temperature difference:

$$\Delta\epsilon_t(t) = \alpha_{th}\Delta T(t) \qquad (7)$$

Where α_{th} is the coefficient of thermal expansion [K^{-1}]; ΔT is the temperature change during the time increment [K].

4) Shrinkage strain

Shrinkage strain can directly be expressed as a swelling strain analogously to thermal strain.

III. MATERIAL PROPERTIES

While the different material parameters used in §II can all be determined experimentally, experimental data are often not available at design stage or sometimes at all to the engineer. Empirical relations are thus needed in order to determine these material properties.

A. Concrete

1) Mechanical properties

In this paper, empirical relations from the fib Model Code 2010 are used [2]. Most concrete properties can be derived from the concrete characteristic compressive strength at 28 days f_{ck} [MPa], the cement type and the type of aggregates. These data are usually all known at design stage.

a) Compressive strength

The development of compressive strength of concrete with time can be expressed as:

$$f_{cm}(t) = \beta_{cc}(t) f_{cm} \qquad (8)$$

Where $f_{cm}(t)$ is the mean compressive strength [MPa] at age t [days]; f_{cm} is the mean compressive strength at 28 days ($f_{cm} = f_{ck} + 8$ [MPa]); β_{cc} is the maturity function defined as:

$$\beta_{cc}(t) = \exp\left\{s\left[1-\sqrt{{}^{28}/_{t}}\right]\right\} \qquad (9)$$

Where s is a coefficient which depends on the type of cement.

b) Tensile strength

The development of tensile strength of concrete with time can be expressed as:

$$f_{ctm}(t) = \beta_{ct}(t) f_{ctm} \qquad (10)$$

Where $f_{ctm}(t)$ is the mean tensile strength [MPa] at age t [days]; f_{ctm} is the mean tensile strength at 28 days ($f_{ctm} = 0.3(f_{ck})^{2/3}$ [MPa]); β_{ct} is the maturity function defined as:

$$\beta_{ct}(t) = \begin{cases} \beta_{cc}(t) \text{ for } t \le 28 \text{ [days]} \\ [\beta_{ct}(t)]^{2/3} \text{ for } t > 28 \text{ [days]} \end{cases} \qquad (11)$$

c) Elasticity modulus

The development of the elasticity modulus of concrete with time can be expressed as:

$$E_c(t) = \beta_E(t) E_c \qquad (12)$$

Where $E_c(t)$ is the elasticity modulus [MPa] at age t [days]; E_c is the elasticity modulus at 28 days defined as:

$$E_c = E_{c,0}\alpha_E\left(\frac{f_{cm}}{10}\right)^{1/3} \text{[MPa]} \qquad (13)$$

Where $E_{c,0} = 21.5\ 10^3$[MPa] ; α_E is a coefficient which depends on the type of aggregates used in the concrete; β_E is the maturity function defined as:

$$\beta_E(t) = \sqrt{\beta_{cc}(t)} \qquad (14)$$

d) Thermal expansion coefficient

The coefficient of thermal expansion depends on the type of aggregates and on the moisture state of the concrete. For the purpose of structural analysis, the thermal expansion coefficient can be taken as:

$$\alpha_T = 10 \cdot 10^{-6}\ [\text{K}^{-1}] \qquad (15)$$

e) Maturity

The effects of cement hydration on the maturity of the concrete can be taken into account using a temperature adjusted concrete age in equations (8), (10) and (13). The adjusted age can be computed as follows:

$$t_e(t) = \sum_0^t \exp\left[\frac{E_a}{R}\left(\frac{1}{T_{ref}} - \frac{1}{T(t)}\right)\right]\Delta t \qquad (16)$$

Where T_{ref} is the concrete placement temperature [K].

2) Thermal properties

a) Density

Reinforced concrete density depends on mix composition (water to cement ratio), density of aggregates, reinforcement ratio and air content. For design purposes, for normal weight concrete, a density of $\rho_c = 2500\ [\mathrm{kg/m^3}]$ can be used.

b) Specific heat

Concrete specific heat depends on the moisture content, maturity as well as the concrete temperature. For low temperatures, it can be taken as:

$$c_p = 900\ [\mathrm{J\ kg^{-1}K^{-1}}] \quad (17)$$

c) Thermal conductivity

Concrete thermal conductivity depends on the concrete temperature and maturity. For low temperatures, it can be taken as:

$$\lambda = 1.9\ [\mathrm{W\ m^{-1}K^{-1}}] \quad (18)$$

B. Cement

Various adiabatic temperature rise curves for different type of cement and concrete aggregates can be found in literature. A simple and effective formula can for example be found in reference [3].

IV. Numerical Analysis

The numerical analysis is performed using the finite element method in two successive stages. First, a transient thermal analysis is performed taking into account the heat generated by the hydration of cement and the heat exchange with the environment and existing constructions. This analysis allows to obtain temperature field in concrete. Secondly, a mechanical analysis is performed to compute strains induced by the temperature field determined at the first stage. The temperature field is also used to determine the time dependent mechanical properties of the concrete.

In the present paper, the method is implemented in the commercial finite-element package ANSYS® [5]. The ANSYS Parametric Design Language (APDL) is used to define the heat generation rate (eq. (3)) at every finite element node for every time step. A user-defined custom material law (Usermat) is implemented for the concrete using the formulation of eq. (4) to solve the structural stresses at every step.

V. Case Study

In order to illustrate the model capabilities, a twelve-meter-long, nine-meter tall and one-meter thick wall is analyzed. It is poured on a previously casted ground slab. The concrete is of class C30/37 according to EN 206-1. The adiabatic heat generation curve is shown on Figure 1. Due to the symmetry of the problem, only half of the wall is analyzed. For simplification purposes, the steel reinforcement is not included. The finite-element model is shown on Figure 2. In the thermal analysis, convective boundary conditions are applied on all external faces of the wall. Ambient temperature and film coefficient are varying to take external temperature and the effects of formwork into account as explained below. On the lower face of the ground slab, a constant soil temperature of 13°C is imposed. In the mechanical analysis, only the lower face of the slab is constrained in every direction.

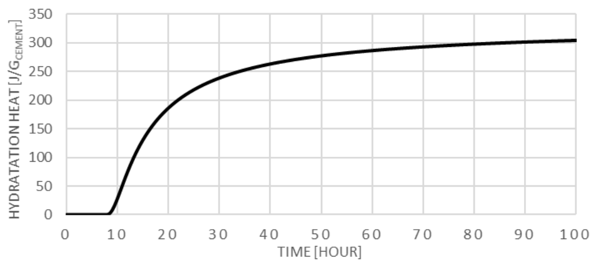

Figure 1. Adiabatic heat generation curve

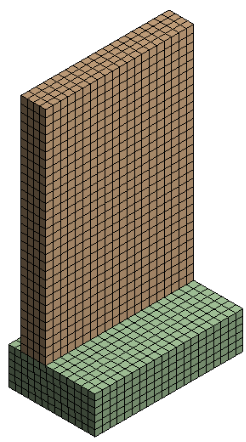

Figure 2. Finite-element model

In this case study, the effect of ambient temperature, placement temperature and curing time (time before formwork removal) on peak temperatures and thermal cracking is investigated. Thermal cracking is estimated by using a cracking index defined as [4]:

$$I_{crack} = \frac{\sigma_1(t)}{f_{ctm}(t)}\ [-] \qquad (19)$$

Where $\sigma_1(t)$ [MPa] is the maximum principal stress acting at time t; $f_{ctm}(t)$ is the temperature adjusted mean tensile strength of the concrete as defined in eq. (10). A cracking index of one or higher means that the state of stress in a region of the concrete can lead to cracking in this region.

The ambient temperature is modelled as a varying sine wave to take lower night temperatures into account:

$$T_{amb}(t) = T_m + \Delta T \sin(2\pi t) \;\; [°C] \qquad (20)$$

Where T_m is the average ambient temperature; ΔT is the temperature delta between maximal and average ambient temperature; t is the time in [days].

The effect of curing time is taken into account by varying the film coefficient on the newly cast wall. The film coefficient can be taken as [4]:

$$h_{wall} = \frac{1}{\frac{1}{h_{air}} + R_{form}} \;\; [\mathrm{W\,m^{-2}K^{-1}}] \qquad (21)$$

Where $h_{air} = 10$ $[\mathrm{W\,m^{-2}K^{-1}}]$ is the air film coefficient; R_{form} $[\mathrm{W^{-1}m^2K}]$ is the thermal resistance of the formwork. In the present study, a 2.7 [cm] thick plywood formwork is considered with a thermal conductivity of 0.13 $[\mathrm{Wm^{-1}K^{-1}}]$. In the absence of formwork, at the end of curing time, eq. (21) reduces to the air film coefficient.

The effect of concrete casting temperature is taken into account by changing the initial temperature of the concrete in the thermal analysis and in eq. (16). For the present study, two different concrete casting temperatures are considered. Moreover, three different ambient temperatures are analyzed with a temperature delta of 5 [°C]. This leads to a total of six cases summarized in TABLE I.

TABLE I. ANALYZED CASES

	1.1	*1.2*	*1.3*	*2.1*	*2.2*	*2.3*
Concrete initial T [°C]	20	20	20	10	10	10
Average ambient T [°C]	20	30	10	20	30	10

The wall is analyzed during the first ten days after casting using a time step of 5000 [s]. The curing is stopped after five days. To highlight the effects of cement hydration and thermal gradients, the effects of creep and shrinkage are neglected in the presented results. This of course leads to artificially high cracking indexes and should not be used for real study cases. The results issued from the numerical analysis are shown on Figure 3. and Figure 4.

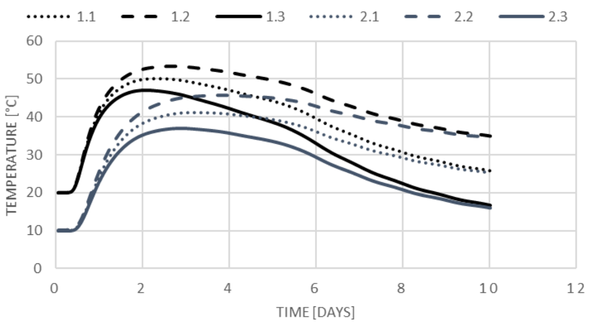

Figure 3. Peak temperature in the wall

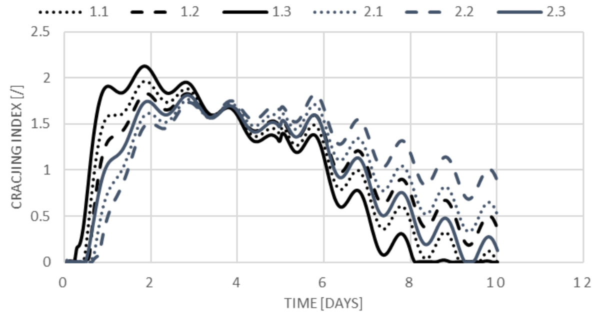

Figure 4. Cracking index at the wall surface

For this specific analysis, the following can be concluded. Concrete should not be placed with a higher temperature than the ambient conditions. Indeed, this leads to a higher cracking index. A lower placing temperature tends to minimize the risk of thermal cracking.

VI. CONCLUSION

A numerical method for analyzing early-age mass concrete was presented. The method consists of a thermal analysis that computes the temperature field developed by cement hydration in the concrete, followed by a structural analysis that allows computing thermal stresses using the previously computed temperatures. The temperature field is also used to determine the time dependent mechanical properties of the concrete. The method can easily be implemented in a finite-element package.

The method allows design engineers to investigate the effect of parameters such as cement class, ambient temperature, concrete placing temperature and curing time on the maximum developed temperature and thermal cracking in the early-age mass concrete. This can allow to reduce the risk of concrete pathologies and thus increasing the lifetime of the constructed facility.

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Metallic Fast Reactor Safety Studies

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I. INTRODUCTION

This presentation will cover the ongoing experimental effort in the frame of safety test for Sodium Fast Reactor (SFR) metallic fuel. The aim of these experiments is to provide the science-based information necessary to reduce uncertainties on transient fuel behaviors. The data collected will be, moreover, relevant to the validation of Modelling and Simulation (M&S) tools. Thus, this work will support the licensing and deployment of future SFR. The experimental program, here described, couples Separate Effects Testing (SET) and Integral Effects Testing (IET), to support the extension of current mechanistic knowledge of fuel behavior and to provide fuel performance data under transient condition. A better description of the program and the experimental plan can be found in [1].

II. EXPERIMENTAL

Three primary activities will be described in this presentation, which will show the IET/SET approach. The SET focused on small-scale experiments and investigated separately the effect of one variable at the time (e.g. microstructure evolution and fission product behavior). Separating these phenomena is necessary to better understand their influence on fuel integral behavior (as described in Fig.1). The IET on the other hand investigate full scale rods in prototypical transient conditions. Several advanced characterization techniques have been used in order to bridge multiscale physics in these studies including: optical microscopy, scanning electron microscopy (SEM) applying energy dispersive spectroscopy (EDS) and EBSD (Electron Back Scattered Diffraction), electron probe microanalyses (EPMA) which uses Wavelength Dispersion X-ray analyses (WDX) and Transmission Electron Microscopy (TEM) coupled with in-situ ion irradiation and heating.

-The first study discussed in this presentation, analyzed the mentioned phenomena after an IET. We analyzed an irradiated fuel rod from an Experimental Breeder Reactor–II (EBR-II) experiment. The Overpower Transient 1 (OPT-1) test in which several pins underwent a moderate reactor ramp.

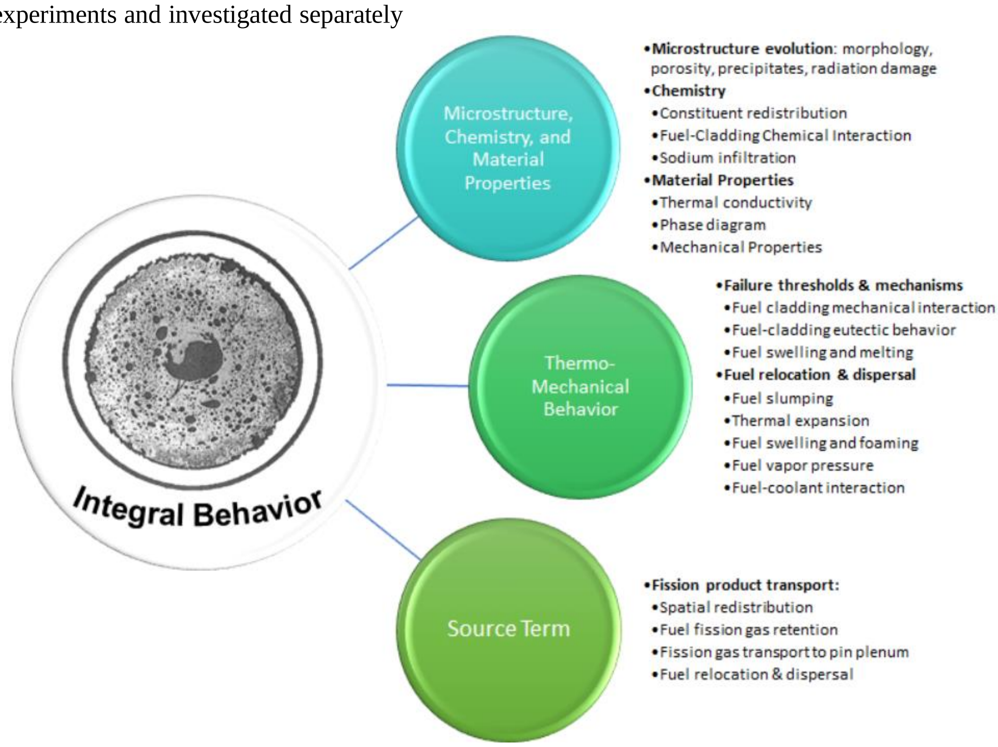

Figure 1. R&D focus areas for fuel safety research for metallic fuel [2].

-The second study, a SET, aimed at separating the influence of different transients (power ramps) on metallic fuel microstructure. We examined the grain structure and porosity of several samples of a simple well characterized metallic fuel alloy (U metal in the orthorhombic alpha phase stable at room temperature).

-The motivation of our final SET study was to evaluate separately the effect of irradiation and temperature on gas diffusion and phase transformation, to understand fission gas behavior in irradiated fuel and in irradiation damage in different phases.

III. Results

A. Irradiated fuel after a transient

A U-Pu-Zr metallic fuel pin, DP-55 X-512, from the EBR-II OPT-1 test X512 (described in [3]), was characterized to understand the impact of an overpower event on fuel microstructure and fission products. Some relevant parameters of this experiment are reported in Table 1, irradiation conditions are described in detailed in [3]. Two sections were chosen from different heights along the irradiated pin for advanced microscopy characterization: 1) a longitudinal cut from the top of the fuel or relevance as failure occurred in the EBR-II design in this location; and 2) a transverse cross section from the middle axial position, important as it can be compare to know steady state irradiation work. The EPMA provided detailed compositional results, which permitted to detect main elements redistribution (U, Pu, Zr) for both samples. Based on these results, three regions could be identified (Fig.2-3) related to different phases formed during irradiation and in line with previous reported behavior for metallic fuel in SFR [4]. Moreover the composition of fission products precipitates was analysed. They consisted of the well established rare earth, Pu and Zr precipitates, in line with past steady state irradiation behavior. Possibly indicating minimal fission product chemistry evolution during the moderate transient. Finally, Fuel Cladding Chemical Interaction (FCCI) was investigated, due to its importance to fuel perfomance as it can lead to cladding failure. Phenomology of FCCI is not well understood, however theory based on past experience is discussed in ref.[5]. On both samples FCCI was observed (Fig.2) to extend to a maximum of 50 μm into the cladding (where lanthanides and Pu migrated) and up to 100 μm into the fuel (with the primary diffusion of Fe and Ni) in line with steady state irradiation behavior. No failure points (e.g. eutetic melt or cracks) were observed in the FCCI region.

These analyses support the conclusion that the samples behaved in line with steady state irradiation of metallic fuel (e.g. [6]), indicating that the moderate transient ramps did not influence fuel performance.

TABLE I Data on X512 experiment, as obtained from [3].

Steady state	**Pin DP-55, X512**	
	Fuel	***Cladding***
11.09% at Burn Up	U-19Pu-10Zr	HT9
Fabrication Injection casting.	Fuel solid slug	75% smear density
Transient	**Ramp**	**T.max center**
32% overpower	5 min at 0.1%/sec	862 °C

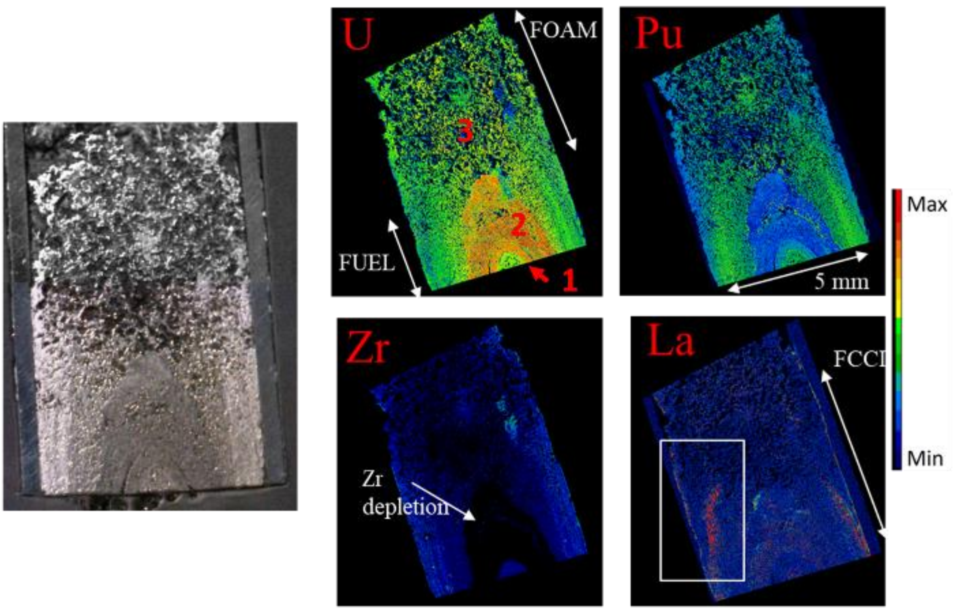

Figure 2. Left optical image of the top of the fuel, right elemental WDX maps obtained by EPMA, showing the 3 regions, and FCCI formation.

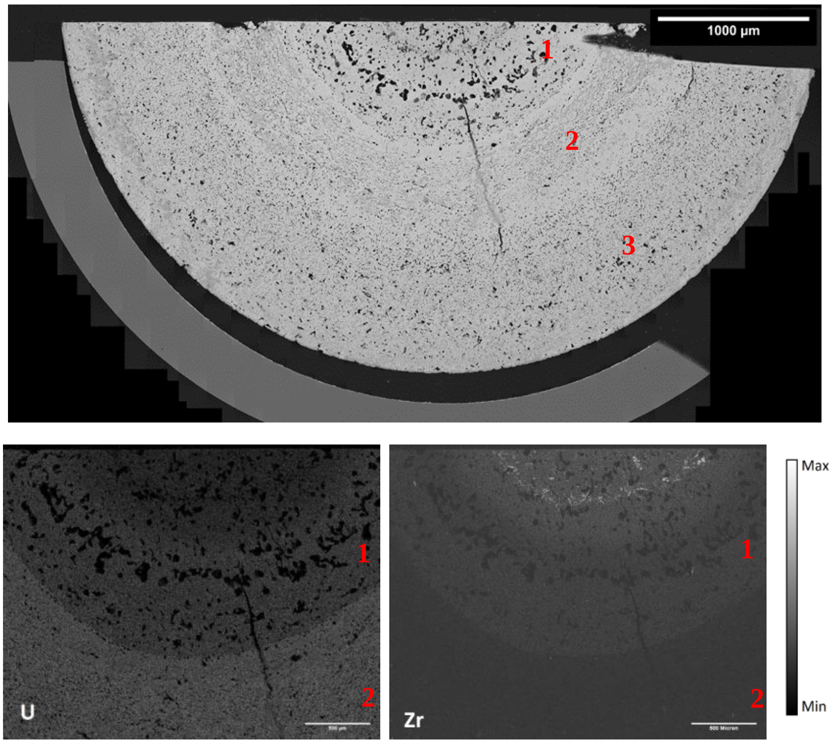

Figure 3. SEM analyses on transverse middle axial position. (Top) SEM Back scattered montage, showing the full cross section analyzed and the 3 main redistribution area (marked with numbers). (Bottom) EDS Maps of the main elements, showing U and Zr element redistribution and Zr precipitates.

B. Grain evolution during transient

This study focused on sheading light on physical mechanisms underlying irradiation growth and swelling in polycrystalline alpha-uranium. The aim of this work was to fill the knowledge gap on early-stage irradiation deformation, which can provide valuable insight into fundamental swelling mechanisms. Such information is especially relevant to physics-based models of microstructure evolution (desired for development of the MARMOT code) and can provide validation to Density functional theory (DFT) simulations. We investigated irradiation-induced grain growth, internal stresses, and plastic behavior by EBSD and SEM analyses. Samples are currently being analyzed and consist of low burn-up irradiated fuel (0.08-0.371 MWd/MTU), which underwent a transient in in the Transient REActor Test facility (TREAT).

C. Gaseous fission product diffusion

The purpose of these experiments was to achieve a deeper mechanistic understanding of fission gas behavior and obtain information on phase transformation by SET testing. This study focused on Xe diffusion, separately evaluating the influence of thermal effects and irradiation-enhanced diffusion on simple and well-characterized samples, with the aim of providing useful information for the development of microstructural based models. By choosing an array of parameters the experiments were designed to separate the influences of radiation damage and temperature on diffusion. The tested specimens (α-uranium and dual phases, α-U/δ-UZr2, U-10Zr) were ion implanted (with Xe) at room temperature and followed by in-situ heating with or without simultaneous ion irradiation (Kr 1 MeV), Table 3 Separate analyses of the defect formation and stability were also performed by ion irradiation with Kr at different temperatures. Heating temperatures were chosen to ensure the stability of the alpha phase and minimize oxidation, while also increasing as high as possible to stimulate diffusion. Alpha-uranium samples showed large grains of ca. 10 µm, while the U-10Zr showed the typical α/δ lamellar structure (Fig. 4), before irradiation. After irradiation, grain restructuring may have occurred for both samples, the bubbles seemed to grow with temperature annealing and even further under Kr irradiation. Further analyses are being performed to quantify bubble density and size their interaction with defects, and to estimate diffusion behavior in the different phases.

TABLE III- EXPERIMENTAL MATRIX APPLIED IN THE IN-SITU TEST.

Sample	**Irradiation Conditions**	**In-Situ heating**	**Irradiation while heating**	**Separate effect**	**Fluence**
	Ion/Energy/ T (°C)	***T (°C)***			***Ion/cm²***
Alpha U-A	Kr/1 MeV/RT	-	No	Defect	$1*10^{16}$
Alpha U-A-1	Kr/1 MeV/250	-	No	Defect	$1*10^{16}$
Alpha U-B	Xe/300 kev/RT	250	No	Temperature effect	$1*10^{16}$
Alpha U-B1	Xe/300 kev/RT	250	1 Mev Kr	Radiation effect	$1*10^{16}$
U-FIB-B2	Xe/300 kev/RT	-	No	Oxidation resistance	$1*10^{16}$
U-10Zr-A	Kr/1 MeV/RT	-	No	Defect	$1*10^{16}$
U-10Zr-A1	Kr/1 MeV/250	-	No	Defect Mobility	$1*10^{16}$
U-10Zr-B	Xe/300 kev/RT	250	No	Temperature effect	$5*10^{15}$
U-10Zr-B1	Xe/300 kev/RT	250	1 Mev Kr	Radiation effect	$5*10^{15}$
U-10Zr-C	Xe/300 kev/RT	500	No	Temperature effect	$5*10^{15}$
U-10Zr-FIB-B2	Xe/300 kev/RT	250	No	Oxidation resistance	$5*10^{15}$

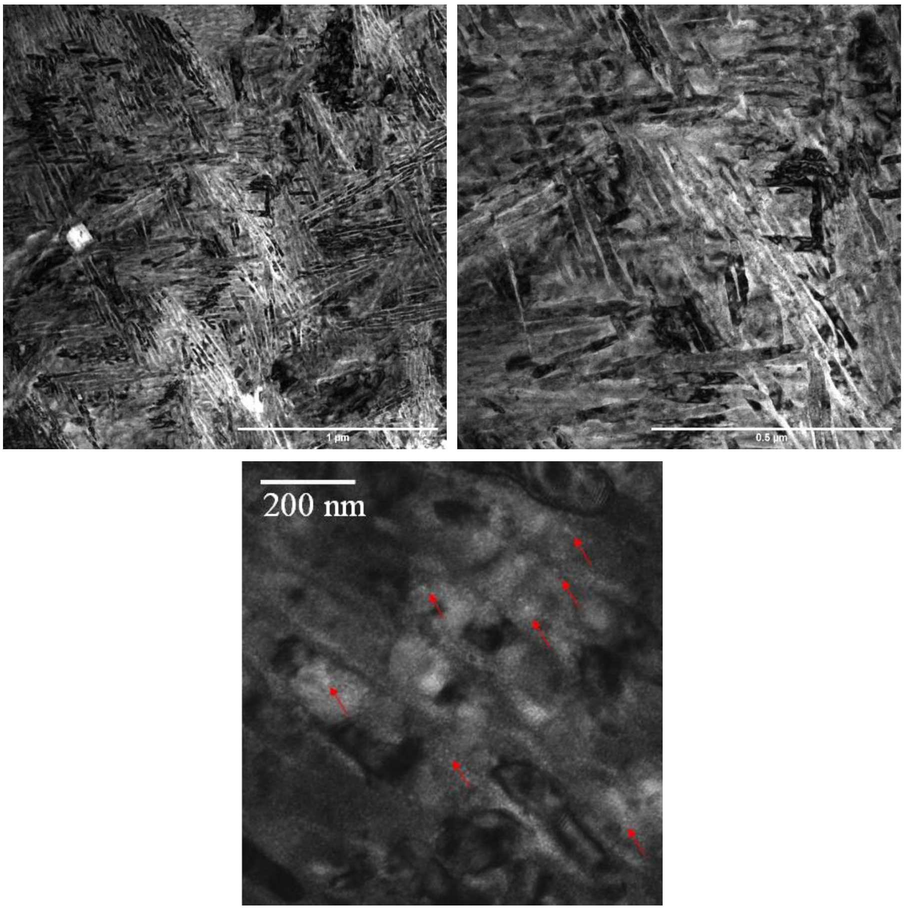

Figure.4 TEM analyses of Uranium Zirconium samples. (Top) Lamellar structure with grains of ca 0.2 - 1 µm in the fresh fuel. (Bottom) Under-focused image showing gas bubble formation at room temperature at $5{\cdot}10^{15}$ ion/cm^2 (U-10Zr-B). The red arrows highlight some of the bigger bright bubbles. Moreover the lamellar structure seems to have evolved under the irradiation and temperature conditions tested.

IV. SUMMARY

A variety of experiments and unique examinations have been presented in this paper. Aimed at analyzing fission products transport in metallic fuel, chemical form of FP after transients; and fundamental microstructural evolution of metallic fuel after a transient. The IET results show that metallic fuel offers a safe performance also under moderate transients. The basic science insight collected from the SETs are relevant to support the development and validation of modelling tools to describe transient metallic fuel performance. These studies will provide important data to understand the behavior of metallic fuel in the future integral test in the TREAT facility.

ACKNOWLEDGMENT

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Engineering MX Precipitation in Ferritic/Martensitic Grade 91 Steel with Wire Arc Additive Manufacturing

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I. INTRODUCTION

The cores of advanced nuclear reactors present extremely harsh environments for materials due to high temperatures, high stresses, chemically aggressive coolants, and intense radiation fluxes [1]. One of the most promising alloys capable of withstanding these environments is the Grade 91 ferritic/martensitic (F/M) steel, a modified 9Cr-1Mo steel with the addition of minor solutes such as V and Nb.

The radiation tolerance and high temperature creep performance of Grade 91 steel is derived from its complex microstructure. In the normalized and tempered condition, its microstructure consists of martensitic laths with an average width of ~0.25-0.5 μm and a dislocation density of 10^{13}-10^{15} m^{-2} [2]. Normalized and tempered Grade 91 also exhibits an array of precipitates, including $M_{23}C_6$ (where M is primarily Cr and Mo) and MX ((V,Nb)(C,N)) precipitates. The formation and stability of MX carbonitrides is necessary for the retention of Grade 91's mechanical properties at elevated temperatures because they pin lath and block boundaries [3]. These particles provide the precipitation strengthening necessary to maintain high creep rupture strength [4], [5], [6]. The MX particles also act as traps for point defects, and hence their radiation stability is also important [4], [7]. Optimal additions of nitrogen during the ingot-making process and optimal heat treatments have been studied to increase MX carbonitride precipitation [8]. However, the resulting components are subjected to creep failure in the heat affected zones of welding [8], [9]. In addition, F/M oxide dispersed steels (ODS) have been developed to increase strength [10], [11], [12]. But manufacturing of complex geometries using ODS methods remains either difficult, costly, or both [8], [13], [14]. Therefore, a need exists to develop an alternative method to traditional metallurgy and to ODS fabrication in order to manufacture large scale, complex geometries of F/M steels with a high density of uniformly distributed precipitates throughout the microstructure for nuclear applications.

Wire arc additive manufacturing (WAAM) presents an opportunity to tailor the microstructures of engineering alloys to obtain desired mechanical properties under elevated temperatures and irradiation. In particular, the composition can be tailored through the optimization of the processing atmosphere, also known as the cover gas. Traditionally, a cover gas is a high purity inert gas such as Ar. The composition of the cover gas influences the chemical reactions that occur in the melt pool during solidification.

This study uses the composition of the processing atmosphere during WAAM to increase MX carbonitride precipitation in the Grade 91 steel during solidification and subsequent heat treatments. This is completed through optimization of the carbon and nitrogen composition in the cover gas. Due to sharp thermal gradients, it is possible to trap solutes such as carbon and nitrogen during the rapid solidification encountered during WAAM and promote precipitation of a fine distribution of carbonitrides. Following WAAM, normalization and tempering treatments were used to homogenize the microstructure.

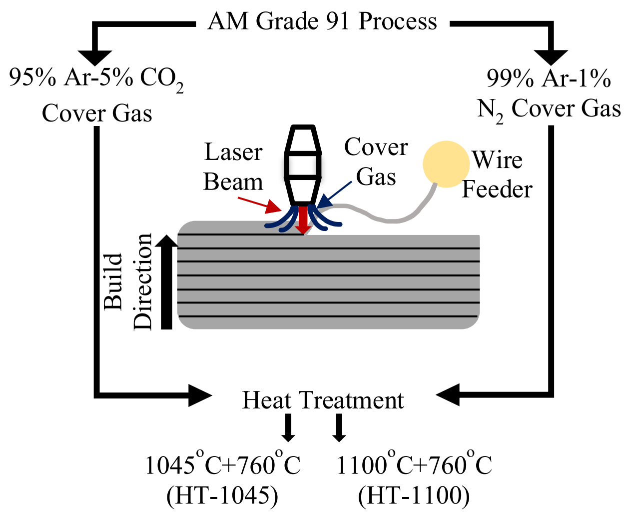

Figure 1. Schematic representation of the wire arc additive manufacturing process and post-processing heat treatment.

The work presented in this paper was supported by the Advanced Fuels Campaign of the Nuclear Technology Research and Development program in the Office of Nuclear Energy, U.S. Department of Energy. The FEI (now Thermo Fisher Scientific) Talos F200X instrument used in this work was provided by the Department of Energy, Office of Nuclear Energy, Fuel Cycle R&D Program and the Nuclear Science User Facilities.

TABLE I. COMPOSITION OF THE WIRE FEEDSTOCK MATERIAL IN WEIGHT PERCENT

	Element															
	Cr	***Mn***	***Mo***	***Nb***	***Ni***	***V***	***Si***	***P***	***S***	***Cu***	***Al***	***B***	***C***	***O***	***N***	***Fe***
Composition	8.62	0.41	0.92	0.08	0.15	0.24	0.31	0.010	0.005	0.05	0.002	0.001	0.08	0.008	0.40	89.07

II. EXPERIMENTAL PROCEDURE

A. Fabrication and Heat Treatment

Two 60x60x10 mm^3 builds were fabricated using a Wolf ABB robot outfitted with a metal inert gas (MIG) welding head. The Grade 91 steel material (Heat 24272) wire feedstock had a diameter of 0.062 inches. The sample fabricated with a cover gas of 95% Ar-5%CO_2 will be designated as Heat 95/5. The sample fabricated with a cover gas of 99%Ar-1%N_2 will be designated as Heat 99/1. Following sample fabrication, two different heat treatments were also completed for each heat to determine the influence of normalizing and tempering on the precipitation in the WAAM builds (Fig. 1). One set of samples were normalized at 1045˚C for 30 minutes and then tempered at 760˚C for 1 hour (HT-1045). The other set of samples were normalized at 1100˚C for 30 minutes and tempered at 760˚C for 1 hour (HT-1100). All samples were air cooled after each heating step. The furnaces used for heat treatment were accurate within 5°C of the target temperatures.

B. Mechanical Testing and Microstructure Characterization

SS-J3 tensile specimens (sheet type dog-bone geometry) with a gauge of $0.76^T \times 1.25^W \times 5^L$ mm^3 were extracted from the Z direction of the as-fabricated build (no heat treatment) [15]. The Z direction is the direction of the build, as shown in Fig. 1. The tensile specimens for this study were tested on a universal load frame at room temperature, 330˚C, and 550˚C in air. All the tensile tests were performed with a strain rate of 10^{-3} s^{-1} using shoulder loading.

Specimens for transmission electron microscopy (TEM) were prepared from the polished tensile specimens using standard focused ion beam (FIB) preparation on a Hitachi NB-5000 FIB/SEM. Characterization via TEM was performed using an FEI (now Thermo Fisher Scientific) Talos F200X scanning transmission electron microscope (S/TEM) operating at 200 kV in ORNL's Low Activation Materials Development and Analysis (LAMDA) laboratory [16]. Energy dispersive x-ray spectroscopy (EDS) was used to map the precipitate distributions [13]. Precipitates were counted and measured using ImageJ from EDS maps. The equivalent diameter (d_{eq}) of each particle was found by measuring the major (a) and minor (b) axes and using

$$d_{eq} = \sqrt{a \cdot b} \qquad (1)$$

Precipitate size distributions were calculated using frequency histograms with built-in Matlab functions (version 2019A).

III. EXPERIMENTAL RESULTS & DISCUSSION

A. Composition

Chemical analysis was performed before and after WAAM to identify the influence of the cover gases on the specimens. Relative to the feed wire composition, the C content increased by 1.3 wt. % in the Heat 95/5 with little variation in the C content for the Heat 99/1. Conversely, the N uptake in the Heat 95/5 was

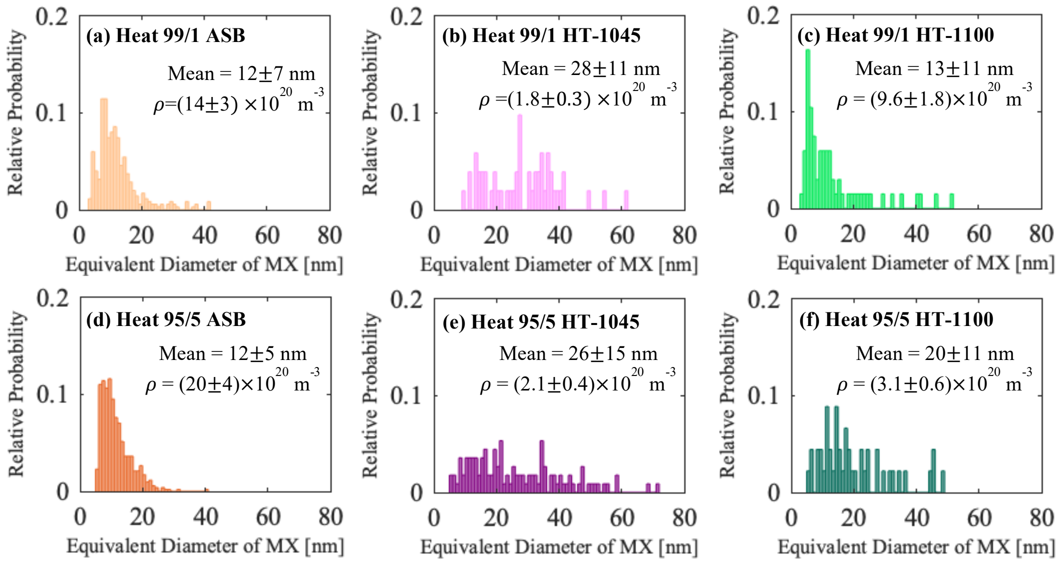

Figure 2. MX size distributions depicted via histograms. The ASB, HT-1045, and HT-1100 conditions of Heat 99/1 are shown in (a-c) and of Heat 95/5 in (d-f).

minimal while a steep increase of 2.48 wt. % is seen in the Heat 99/1, relative to the feed wire composition.

B. Microstructure Characterization

The as-built specimens of Heats 95/5 and 99/1 displayed a refined columnar grain structure oriented in the build direction. This grain structure forms as a result of the thermal gradient [17]. Both heat treatments caused significant prior austenite grain size reduction from the as-built specimens of both heats and homogenized the grain morphology. The heat-treated specimens showed a tempered martensitic lath structure (Fig. 3).

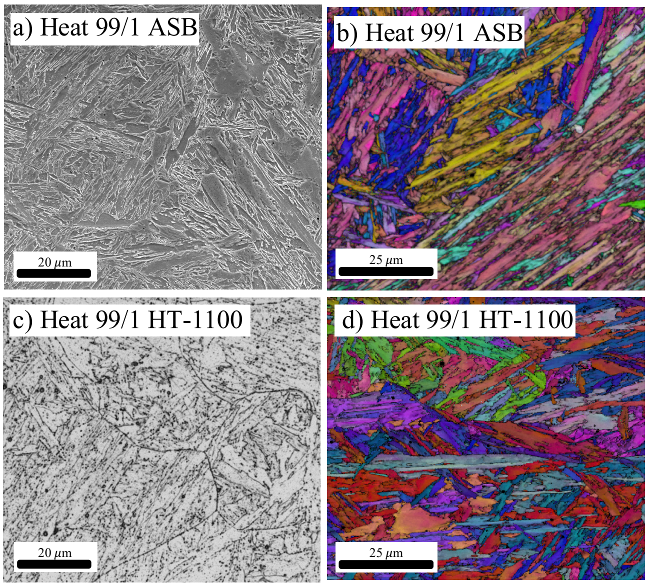

Figure 3. (a) SEM micrograph and (b) EBSD image for the Heat 99/1 as-printed specimen. (c) SEM micrograph and (d) EBSD image for the Heat 99/1 HT-1100 specimen. These are qualitatively representative of the Heat 95/5 specimens.

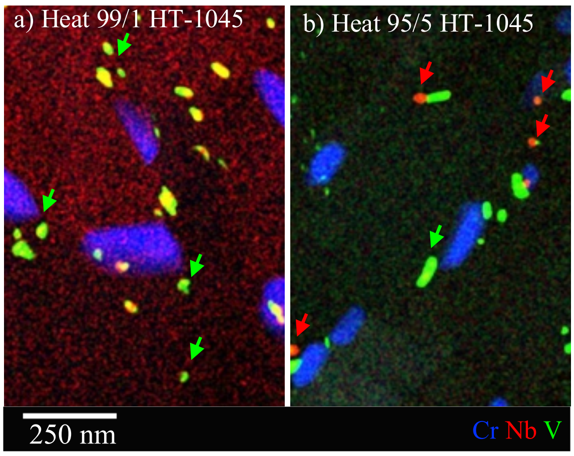

Figure 4. STEM-EDS count generated color overlap maps for precipitate elements in the HT-1045 conditions of (a) Heat 99/1 and (b) Heat 95/5. The green arrows point to examples of V-rich MX precipitates and the red arrows point to examples of Nb-rich MX precipitates.

Both as-built heats display disk-shaped carbonitrides enriched in Nb and V ((V,Nb)(C,N)) on and within the prior austenite grain boundaries (PAGBs). However, only Heat 95/5 exhibited Nb(C,N) particles on PAGBs in all conditions. These Nb(C,N) particles helped to pin grain boundaries, resulting in ~15 μm smaller prior austenite grains post-heat treatment in Heat 95/5 versus Heat 99/1. In comparison, the N-rich Heat 99/1 did not exhibit Nb-rich particles. Instead, increased N content qualitatively increased the population of V(C,N). This is consistent with previous work that shows N content is the dominant factor for the driving force of VN nucleation [8].

Both as-built conditions precipitated a high density of small (~12nm) (V,Nb)(C,N) particles on the order of ~10^{21} m^{-3}. This density is 2-3 orders of magnitude greater than the MX density in traditionally processed Grade 91[18], [19]. A high density of precipitates has been shown to increase a material's sink strength, as those precipitates act to annihilate point defects. The increased carbonitride density created by WAAM enhanced Grade 91's sink strength by an order of magnitude (Fig. 5). Sink strength was calculated from Refs. [20] and [21] assuming that the semi-coherent MX particles act as coherent particles. Future studies are needed to determine the stability of these precipitates under thermal aging and irradiation.

IV. Mechanical Characterization

Yield strength, ultimate tensile strength, uniform elongation, and total elongation were measured for each condition. The as-built conditions of Heats 99/1 and 95/5 displayed greater yield and ultimate tensile strengths than the heat-treated conditions. The tensile and total elongation data are graphically represented in Fig. 6. The heat treatment needs

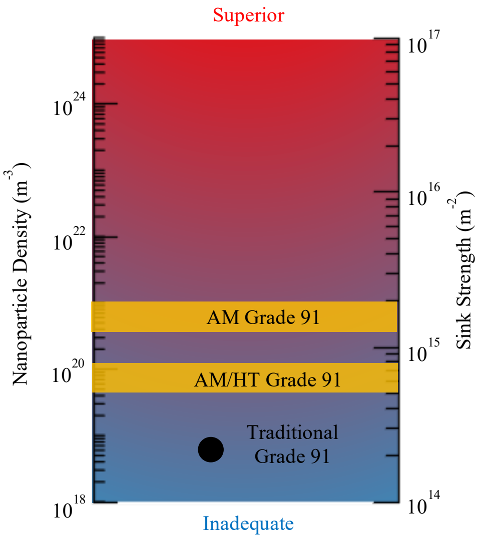

Figure 5. Sink strength results for WAAM fabricated Grade 91 (AM Grade 91), heat treated WAAM fabricated Grade 91 (AM/HT Grade 91), and traditional Grade 91.

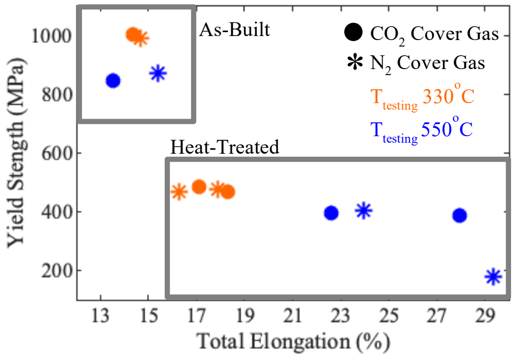

Figure 6. Yield strength and total elongation results for WAAM fabricated Grade 91 from tensile testing at 330°C (orange) and 550°C (blue). Results for samples fabricated in CO_2 cover gas are demarcated with dots and those fabricated in N_2 cover gas are demarcated with asterisk marks.

further optimization to maintain the high strength of the as-built specimens.

V. Conclusion

Ferritic/martensitic steels such as Grade 91 can suffer from low creep rupture strength at elevated reactor temperatures. A solution can be engineered by creating a high density of MX precipitates. These particles also act as sinks that serve as sites for defect annihilation. In this study, manipulation of the cover gas concentration of C and N during WAAM induced a carbonitride density two orders of magnitude higher than traditionally processed Grade 91. This study proves WAAM to be a promising step forward and a powerful tool in engineering microstructural features in ferritic/martensitic steels.

Acknowledgments

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Utilization of Miniaturized Testing Specimens for the Irradiated Reactor Pressure Vessel Materials Degradation Evaluation

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I. Introduction

Nuclear energy is one of the most important sources of energy in the Czech Republic. The operation of two Nuclear Power Plants (NPP) with a total of 6 units requires periodic assessment of safe and long–term operation, especially with the connection to current trends of components operational lifetime extension. In the Czech Republic eastern types of Pressurized Water Reactors (PWR) known as WWER developed in former Soviet Union are operated. Type WWER–440 is used in NPP Dukovany (4 units) and WWER-1000 in NPP Temelín (2 units). In the frame of Reactor Pressure Vessel (RPV) surveillance programs the support to NPP is provided by ÚJV Řež, a. s. (Nuclear Research Institute). The RPV is considered as a non-replaceable reactor component, so it is essential to ensure the strength and integrity of the RPV materials for reliable and safe operation of the nuclear reactor. The pressure vessel material is exposed to long-term intense neutron flux, thermal and pressure changes. It is therefore necessary to monitor the degradation rate of the material to determine the residual lifetime.

The Integrity and Technical Engineering Division consists of several departments and one of them is the Mechanical Testing Department. It includes an accredited hot cell testing laboratory that offers a wide range of mechanical testing methods of irradiated materials with the portfolio of 21 accredited test methods.

A system of 51 hot and semi-hot cells located on three floors is a unique experimental capacity in Central Europe (Figure 1). Hot cells are shielded by 1 m of heavy concrete and the assumed activity is up to 370 TBq. Semi-hot cells are shielded by 150 mm of lead with the processable activity up to 37 GBq.

Considering the limited availability of original archive materials, new perspective methods of testing are being implemented in the process of irradiated RPV materials testing (e.g. Small Punch Test (SPT), miniature CT specimens). This paper describes the process of necessary operations and modifications of equipment in the hot cell laboratory to enable the implementation of new methodologies of neutron irradiated sub-sized specimens testing for RPV materials degradation evaluation with a special attention to the SPT methodology.

Figure 1. Accredited Mechanical Testing laboratory, ÚJV Řež a. s.

II. Test methods on sub-sized specimens

The purpose of use the miniaturized specimens is to reduce the required volume of testing material and to enable testing of materials of operated industrial components. Large test specimens are used in conventional mechanical properties test methods and sampling of sufficient material volume is usually complicated without influence on the evaluated component operation and integrity.

A. Small Punch Test (SPT)

One of the most useful miniaturized specimens test method is SPT that is based on the testing of thin clamped specimens (usually with a diameter of 8 mm and thickness of 0.5 mm). During the test a puncher of specific dimensions deform the specimen surface and a force-deflection or force-displacement curve is obtained (Figure 3). Configuration of the testing apparatus shows Figure 2.

The purpose of the SPT is to determine the parameters for estimation of evaluated material mechanical properties (yield strength, ultimate tensile strength or fracture mechanics properties). To obtain material for the testing, established and proven sampling methods are used, e.g. electric discharge machining or milling with hemispherical cutter. Another possibility of sample preparation is the utilization of already tested standard irradiated specimens from surveillance programs.

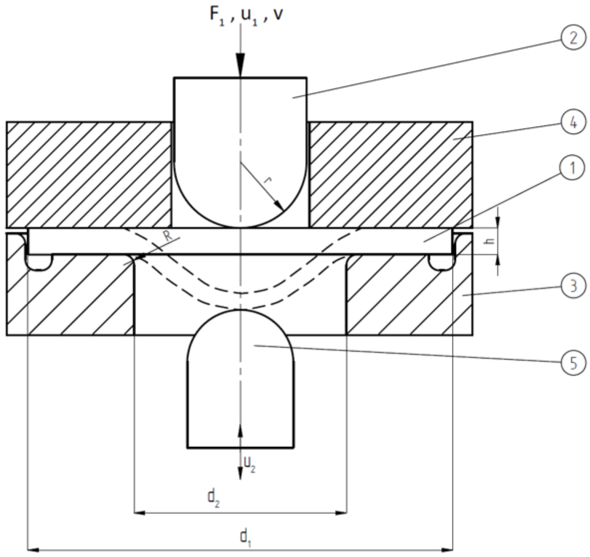

Figure 2. Configuration of the SPT apparatus (1 - specimen, 2 - punch, 3 - receiving die, 4 – clamping die, 5 – deflection measurement rod) [1]

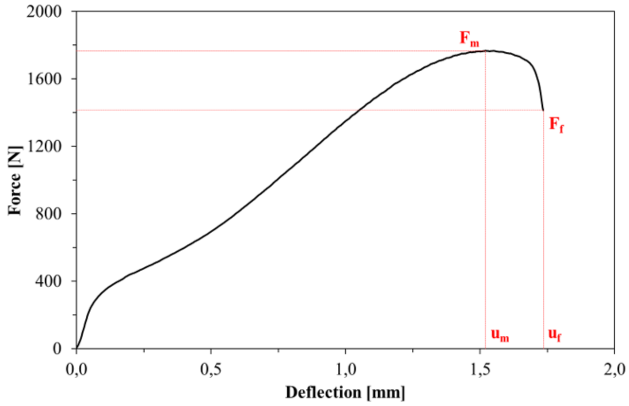

Figure 3. Force-Specimen deflection curve recorded during a SPT of a ductile material [1]

B. *Automated Ball Indentation Test (ABIT)*

ABIT is considered as a semi-destructive testing method and can be used for the determination of tensile properties of operated components using in-situ measurements or testing in the laboratory conditions. The devices for this test method, (usually the electro-mechanical testing machines), are equipped with the selected indenter of various diameters (2.5; 1.575; 0.762; 0.508 mm), that is made from tungsten carbide.

By multiple loading and subsequent unloading of the indenter by 50 % in gradual steps, the force-displacement of the indenter is obtained. Typical representation of data record obtained during ABIT is shown below in Figure 5.

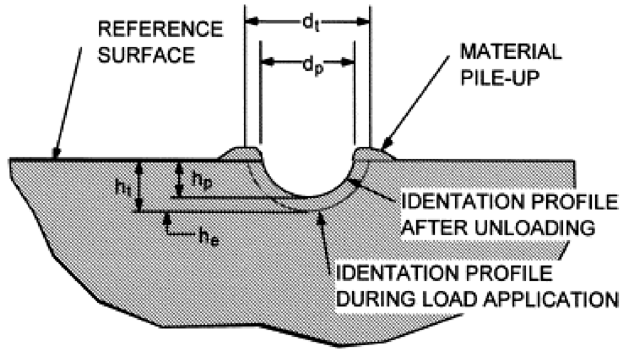

Figure 4. ABIT indent geometry after force removal (complete unloading) [2]

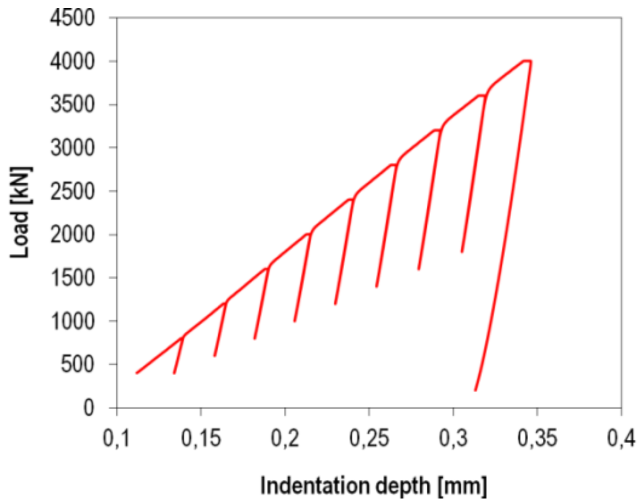

Figure 5. The loading diagram of ABIT of A533B (JRQ) in irradiated state (24 °C, indenter 2,5 mm, fluence $12.5 \cdot 10^{22}\,m^{-2}$)

C. *Miniaturized CT specimens*

Mini-CT (Compact Tension) specimens (Figure 6) are used for static fracture toughness testing. Currently the use of this miniaturized specimen geometry is usually connected with the machining from the tested irradiated Charpy type specimens for the significant results database enlargement. Two tested Charpy type specimens provide enough volume of material for preparation of 16 mini-CT samples.

Testing of the mini-CT sample is carried out in the same way as with the conventional CT fracture toughness specimen test. During the test, parameters like temperature, load and crack opening displacement are recorded. The force is recorded by a load cell, crack opening displacement by a clip gage.

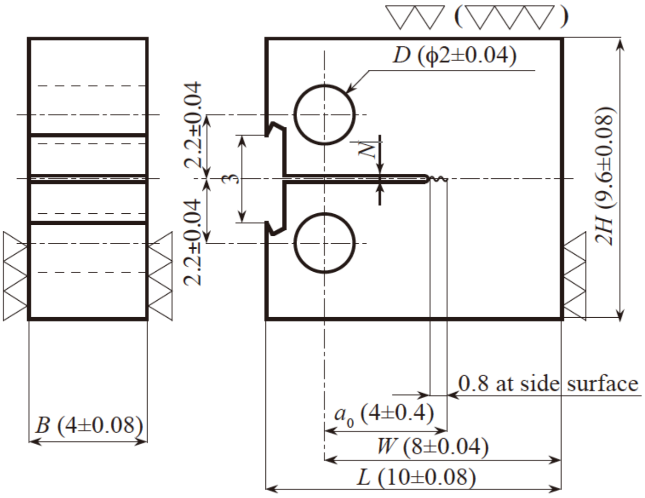

Figure 6. Miniaturized CT specimen dimensions [3]

D. *Implementatation of SPT methodology for irradiated materials testing*

Implementation of SPT methodology was connected with necessary modifications in Hot Cell facility procedures and production and testing equipment. The existing equipment has been modified (EDM) and new equipment was installed (specimen polishing devices, specimen holders).

The samples for SPT are currently produced by EDM. For specimen preparation from smaller volumes of material (e.g. sampling from miniaturized broken Charpy specimens) metallographic diamond saw is used.

To achieve the required accuracy for the test specimen thickness h_0, additional equipment consisting of the automatic surface polisher was developed. After the initial testing, the equipment for polishing was installed in a Hot Cell (Figure 7 and Figure 8), combined with supporting instrumentation and modified master-slave remote control system. The system design provides test specimen preparation in accordance with requirements for the SPT specimens [1], [2], [4].

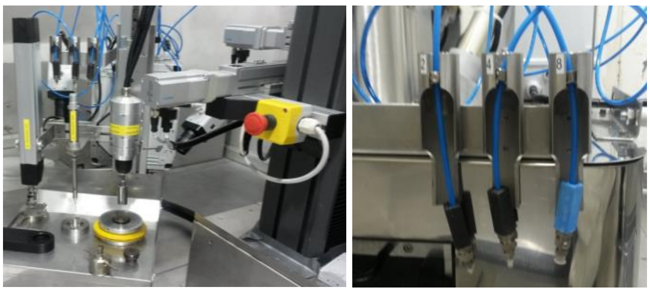

Figure 7. Equipment for assembly of the specimens testing fixtures (left) and the system of pneumatic specimen holders (right)

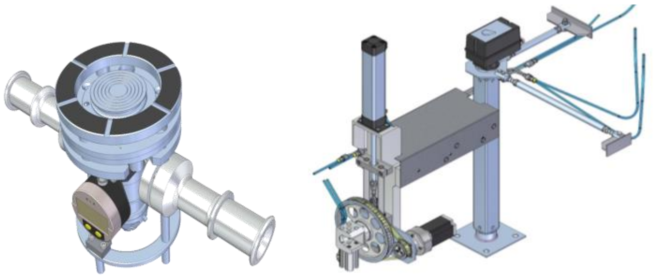

Figure 8. New polishing equipment installed in the hot cell - specimen holder and robotic arm

Expected problems with sample handling and manipulation were solved with the introduction of specimen pneumatic holders. Every Hot Cell involved in the SPT testing was equipped with the system of pneumatic sample holders (Figure 7) to prevent the possible damage of the thin specimen with the grip of manipulator.

SEM has been also modified for the realization of fractography analyses of the fracture surfaces on irradiated SPT specimens. In the SPT, it is important to determine a fracture mode and document the absence of any fracture anomalies during the evaluation of the obtained test results (Figure 9). SEM analyses can be also advantageously used for the evaluation of the fracture mechanics parameters from the SPT results [5].

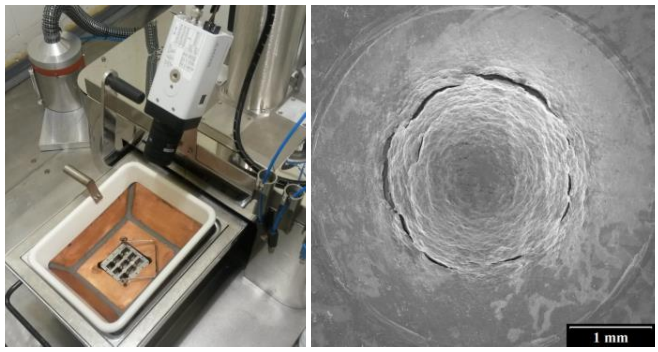

Figure 9. SEM modifications for SPT specimens (left) and example of fracture surface on SPT specimen (material JRQ) (right)

E. *Comparison of SPT with conventional methods*

To confirm the suitability of the SPT and ABIT methodology for the determination of mechanical properties of irradiated structural materials of nuclear reactors, the verification program has been performed. Results of conventional uniaxial tests were compared with the evaluated tensile data from SPT and ABIT methods. As a selected material, the IAEA correlation material A533B (JRQ) [6] was used for the purposes of the verification program in initial and irradiated state (neutron fluence $12.5 \cdot 10^{22}$ m^{-2}, E>0.5 MeV). Material is well described and is also used as a reference material in surveillance programs of the Czech NPP for the documentation of irradiation conditions stability.

The tensile testing specimens were prepared with the geometry of 4 mm diameter and 20 mm gauge length in the transverse orientation in original block of material 5JRQ52. Tests are carried out at room temperature and +265 °C in accordance with the ISO 6892 [7]. Video extensometer was used for precise measurements of specimens' elongation. For each testing temperature an material condition three tensile were used. ABIT and SP tests were performed in the identical temperature range. Using the current pre-normative documents [1], [2], tensile data from ABIT a SPT were evaluated and correlations were established (Figure 10 and Figure 11). In initial state very good agreement of SPT and ABIT results was obtained, however in initial state higher scatter of results was observed. Improvement of correlations precision will be achieved by the increase of the tested samples number and this adjustment will be implemented in the next phases of the verification program at the hot cell facility.

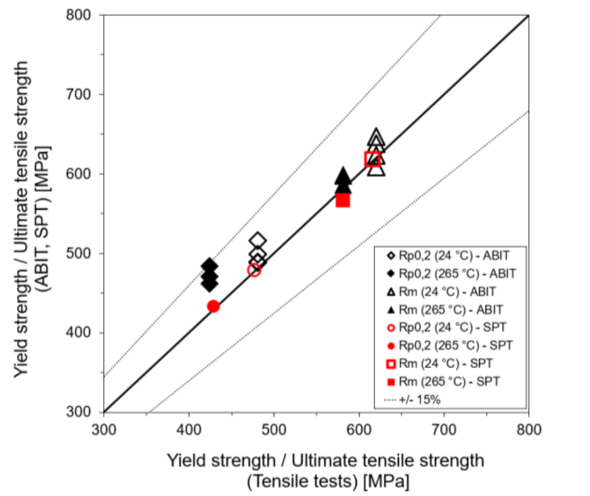

Figure 10. Correlation of results from SPT, ABIT and uniaxial test results (unirradiated material A533B - JRQ) [8]

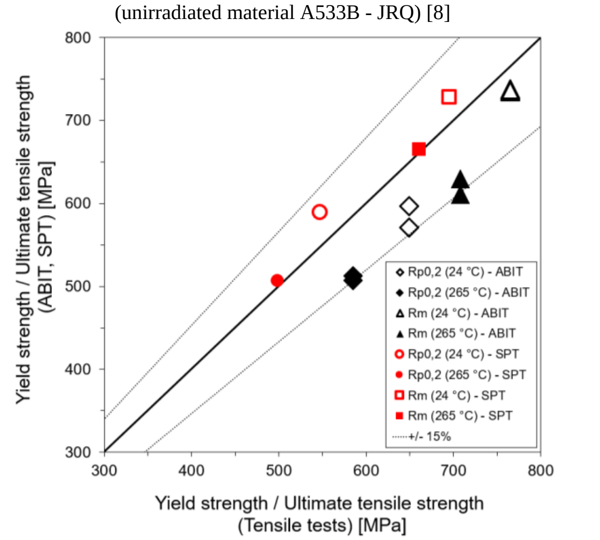

Figure 11. Correlation of results from SPT, ABIT and uniaxial test results (irradiated material A533B - JRQ) [8]

III. Conclusion

Perspective methods for the miniaturized specimens testing and their importance for the irradiated RPV materials degradation evaluation was described and subsequently verified by the first phases of the experimental program on the A533B (JRQ) material. Obtained results have shown the usability of the SPT and ABIT methods for the determination of mechanical properties of the neutron irradiated structural materials. Use of these methods will bring the lower consumption of archive NPP materials, enlarge the volume of irradiated structural NPP materials experimental results database and will also reduce the dose obtained by the personnel of the Hot Cells testing laboratories. Evaluated results will be also used in the ongoing SPT standardization activities in ASTM and EN [8], where ÚJV Řež, a. s. actively participates. Fully fledged international standard is expected to be published during the 2020.

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Development of ATTILHA (Advanced Temperature and Thermodynamics Investigation by a Laser Heating Approach) set up for high temperature applications

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I. Introduction

Thermodynamic and thermophysical properties of high temperature liquid (T>2000 K) are paramount data for the modeling of high temperature industrial processes. Nevertheless, many industrial applications lack basic data on solid and liquid materials.

To meet the needs of these industries (aerospace, nuclear, glass and ceramic, additive manufacturing) the ATTILHA (Advanced Temperature and Thermodynamics Investigation by a Laser Heating Approach) set up has been developed.

In the proposed poster, the different configurations (levitation, self-crucible) of the device are presented. Some experimental results obtained with both configurations are also presented.

II. Experimental Setup

ATTILHA can be used in two different configurations depending on the experimental data that one is interested in. In both configurations, the sample is heated by means of a CO_2 laser emitting at 10.6 µm. A bi-chromatic pyrometer, an infrared detector (optional) and an infrared thermal camera monitor the temperature. Also, the camera allows to observe temperature gradients and emissivity variation during the experiments. An ultra-fast camera (f > 1 kHz) can also be used to follow the evolution of the shape of the samples during the solid to liquid transitions. In the following sections, the aerodynamic levitation and the containerless configurations will be described.

A. Aerodynamic levitation setup

In this configuration, the sample has a spherical shape and levitates in a stream of gas (reducing or oxidizing) out of an aluminum nozzle. The pressure and the gas flow can be adjusted to optimize the quality of the levitation.

A schematic view of the experimental setup is reported in Fig. 1.

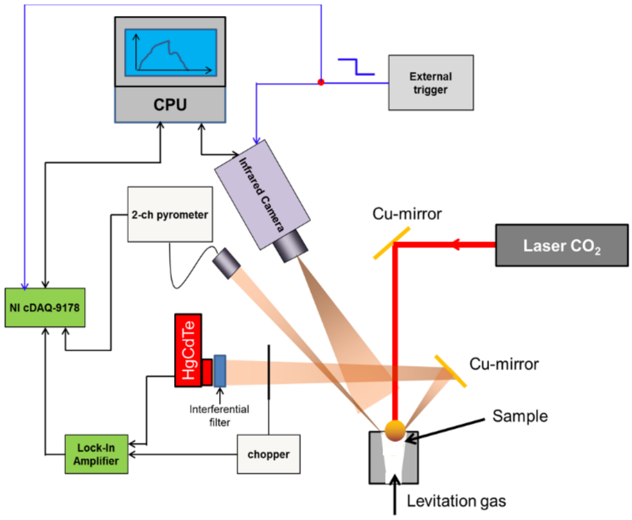

Figure 1. Schematic view of the ATTILHA setup in aerodynamic levitation configuration [1].

The advantage of this configuration is that the sample is not in contact with a crucible, avoiding any high temperature chemical reaction, which may alter the composition of the samples itself. However, a temperature gradient may be present across the sample.

B. Containerless setup

In this configuration, the sample is placed inside a steel vessel and maintained using three ceramic screws. The advantage of this technique is that the laser heats only the central part of the sample. Therefore, the remaining sample surrounds the heated surface leading to a so-called "self-crucible" configuration, in which the crucible is the sample itself. This setup allows limiting the interaction of the heated zone of the sample with any component of the vessel, avoiding any chemical contamination from the surrounding materials. A schematic view of the experimental setup is shown Fig. 2.

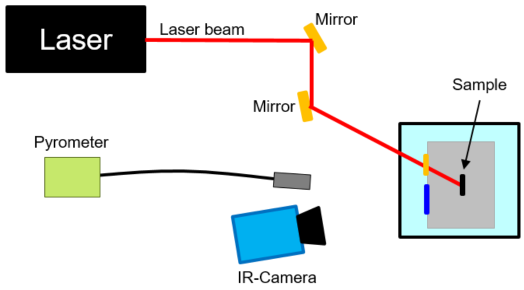

Figure 2. Schematic view of the ATTILHA setup in contactless configuration.

This configuration allows studying samples containing uranium.

III. Results

A. Aerodynamic levitation results

Experimental results have been obtained on the Al_2O_3-ZrO_2 system. The aim of this experimental campaign was to confirm the eutectic composition and temperature of this system. Both the pyrometer and the infrared detector have been used to measure the temperature of the heated sample (Fig. 3). Two thermal arrests are visible on the cooling flank recorded by the bi-chromatic pyrometer, whilst the infrared detector recorded only one inflection (2250±96 K). The inflection at higher temperature corresponds to the liquidus transition. The measured value is in agreement with the literature data [2] and the calculated phase diagram (Fig. 4). The second inflection recorded by the bi-chromatic pyrometer is the eutectic transition, again in good agreement with the literature.

On the heating flank, the temperature measured by the infrared detector and the pyrometer differs significantly. The pyrometer recorded a constant temperature. The infrared detector measured a rising temperature until the laser was switched off. This difference is related to the wavelength of measure for the pyrometer (around 1 µm) and for the infrared detector (10 µm): the optical properties of the sample are significantly different at the two wavelengths, leading the pyrometer to integrate the radiant flux coming from the surface as well as from a certain amount of the liquid sample underneath. In these conditions, the temperature measurement of the bi-chromatic pyrometer cannot be considered as only representative of the surface conditions, but as the combination of both surface and bulk contributions.

Current experimental results are in good agreement with the available phase diagram [2]: the measurement confirmed the eutectic temperature as well as the eutectic composition (around 31 mol% ZrO_2).

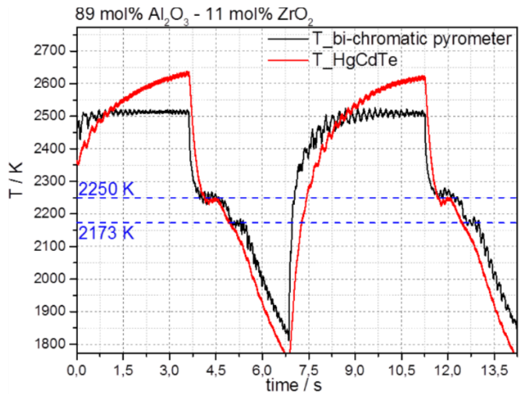

Figure 3. Thermograms obtained using a pyrometer (black line) and an infrared detector (red line) on an Al_2O_3-ZrO_2 sample.

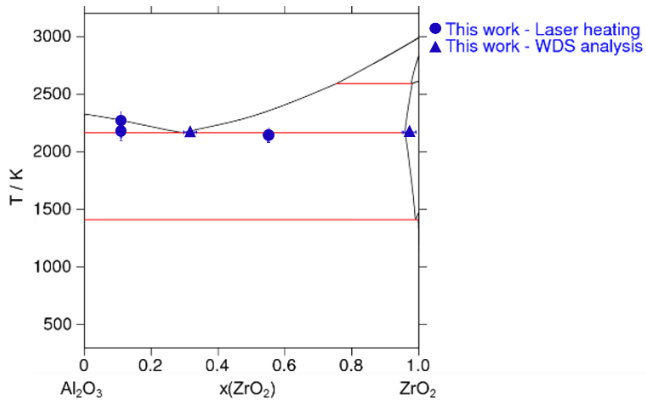

Figure 4. Comparison between current experimental results and calculated Al2O3-ZrO2 phase diagram. Phase diagram has been calculated using the TAF-ID database [3]. Red lines on the diagram correspond to invariant transitions, such the eutectic reaction around 2173 K. WDS=Wavelength-dispersive X-ray spectroscopy.

B. Containerless experimental results

An example of experimental output obtained using the containerless configuration is given in Fig. 5 on an UO_2 pellet heated under Ar. Performing the heat treatment under Ar allowed to reach congruent vaporization [4,5].

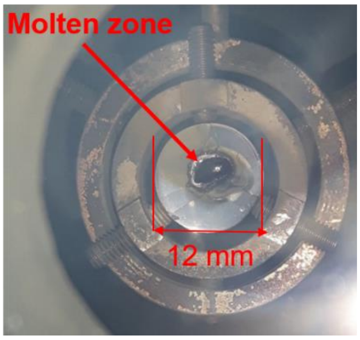

Figure 5. UO_2 pellet. The center of the sample was molten using the laser heating technique. It can be noted that the molten region is surrounded by solid UO_2, avoiding any interaction with the maintaining screws.

The corresponding thermogram obtained using the pyrometer is reported in Fig. 6. After a pre-heating stage at around 1500 K (in order to reduce thermal-mechanical stresses in the pellet [6,7]), power of the laser has been set to overcome the expected melting temperature value by around 200 K. This allowed to form enough molten material to be observed by the pyrometer. Heating rate has not been measured.

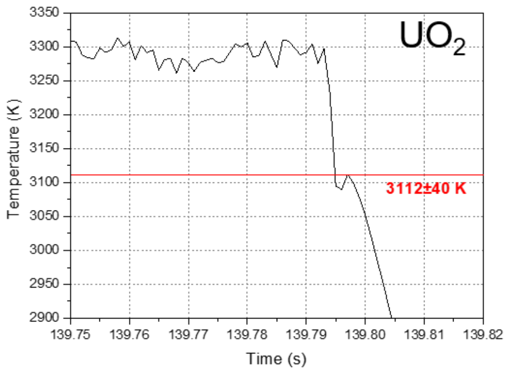

Figure 6. Thermogram obtained on a UO_2 sample.

The measured melting temperature (3112±40 K) is in good agreement with the literature value [8]. Energy Dispersion Spectroscopy (EDS) did not reveal any significant change in the stoichiometry of the molten sample. However, this technique does not allow to have fully quantitative results on light elements such as oxygen. The fact that the melting temperature measured during the present work in consistent with previous experimental results obtained with different techniques (see for example [4,5,9]) may indicate that after-test stoichiometry was not far away from the initial O/U=2 value.

IV. Conclusions

ATTILHA is a versatile experimental setup, which allow high temperature measurements on different kind of materials; ceramic, metallic as well as oxide-metallic samples can be investigated. Both aerodynamic levitation and self-crucible conditions can be used. Preliminary experimental results on the Al_2O_3-ZrO_2 system confirmed the eutectic temperature and the eutectic composition with the literature. Using the self-crucible configuration, a UO_2 pellet has been studied showing a good agreement with the literature.

In the future, this setup will be used to investigate the liquid miscibility gap in the U-Zr-Fe-O system in the frame of the severe accident studies on PWR. Furthermore, Comsol ® multi-physics calculations will allow to optimize the experimental condition used during the aerodynamic levitation experiments.

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TRACK 6: NUCLEAR SAFETY, SECURITY AND RADIATION PROTECTION

GENERIC PROBABILISTIC SAFETY ASSESSMENT MODELS FOR INTERNATIONAL PRECURSOR ANALYSIS APPLICATIONS

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SYSTEMATIC APPROACH TO TRANSBOUNDARY ATMOSPHERIC DISPERSION ASSESSMENT OF A HYPOTHETICAL RELEASE FROM A NUCLEAR POWER PLANT

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UTILIZATION OF ARTIFICIAL INTELLIGENCE IN THE ANALYSIS OF NUCLEAR POWER PLANT REQUIREMENTS

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MEASUREMENT OF SCINTILLATION LIGHT YIELD IN NAI(TL) DETECTOR FOR THE SABRE DARK MATTER EXPERIMENT

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DECOMMISSIONING A FUEL ASSEMBLY MANUFACTURING PLANT IN BELGIUM: HEALTH PHYSICS ASPECTS FROM FIRST METHODOLOGY TO FINAL RELEASE

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EXPORT CONTROLS IN THE SUPPLY CHAIN OF NUCLEAR PRODUCTS AND TECHNOLOGIES. THEORY AND PRACTICE

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APPLICATION OF PROBABILISTIC SAFETY ASSESSMENT TO JUSTIFICATION OF DESIGN SOLUTIONS RELATED TO ULTIMATE HEAT SINK PROTECTION FROM EXTERNAL HAZARDS

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VISUALIZATION OF RADIOACTIVE SUBSTANCES USING FREELY MOVING GAMMA-RAY IMAGER BASED ON STRUCTURE FROM MOTION

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CORRELATION BETWEEN HEAT-MASS TRANSFER, CHEMICAL REACTIONS AND PHASE TRANSFORMATIONS IN CORIUM MELT LOCALIZATION DEVICES DURING SEVERE NUCLEAR POWER PLANT ACCIDENTS

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A STUDY OF DIRECTIONAL GAMMA-RAY DETECTOR WITHOUT SHIELD BY MONTE CARLO SIMULATION

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DEVELOPMENT OF ONE-DIMENSIONAL OPTICAL FIBER TYPE RADIATION DISTRIBUTION SENSING METHOD BASED ON WAVELENGTH SPECTRUM UNFOLDING

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AN ALGORITHM FOR REDUCTION IN COUNT RATE FLUCTUATIONS, IMPROVED RELATIVE STANDARD DEVIATION, FASTER RESPONSE TIME & SPURIOUS REJECTION IN NUCLEAR PULSE COUNTING SYSTEMS

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A COMPARATIVE ANALYSIS OF REGULATORY PROVISIONS FOR ENVIRONMENTAL SAFETY THROUGH THE LIFECYCLE OF NUCLEAR POWER STATIONS IN THE UNITED KINGDOM AND SOUTH AFRICA

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INTEGRATION OF FISSION PRODUCT RELEASE ANALYSIS FROM REACTOR PRESSURE VESSEL AND SPENT FUEL POOL IN MODIFIED ART MOD 2

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MICROWAVE TECHNOLOGY FOR PERSONAL MONITORING AS A NUCLEAR SECURITY APPLICATION

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THE COMPLEXITIES OF LAYERED OBLIGATIONS IN INTERNATIONAL NUCLEAR COMMERCE PEACEFUL USE AGREEMENTS

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OCCUPATIONAL EXPOSURE PROFILE AND ITS IMPLICATION IN THE CURRENT INDIVIDUAL MONITORING PROGRAM OF NUCLEAR MEDICINE WORKERS IN THE PHILIPPINES

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INTEGRATION OF RESEARCH REACTOR COMPUTER SECURITY MANAGEMENT WITH WHOLE-OF-FACILITY ASSET MANAGEMENT PRACTICE

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FLEX STRATEGIES ANALYSIS UNDER A LBLOCA SCENARIO WITH THE MELCOR CODE

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THE ROLE OF NUCLEAR FORENSICS IN SUPPORTING THE PEACEFUL APPLICATION OF NUCLEAR SCIENCE AND TECHNOLOGY: AN AUSTRALIAN PERSPECTIVE

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WHEN RADIATION PROTECTION AND OCCUPATIONAL HYGIENE MEET: A CASE STUDY ON URANIUM

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THE FEASIBILITY OF BLOCKCHAIN APPLICATION TO NUCLEAR SAFEGUARDS IN THE IAEA

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A COLLABORATIVE EXPERIENCE: IMPLEMENTATION OF AN OPTIMISED TC-99M GENERATOR ASSEMBLY PROCESS

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UNDERSTANDING DIGITAL TRUST AND SEGMENTATION IN NUCLEAR FACILITIES

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STUDY OF RADIOACTIVE CORROSION PRODUCTS IN DUKOVANY NUCLEAR POWER PLANT AT CZECH REPUBLIC

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Generic Probabilistic Safety Assessment Models for International Precursor Analysis Applications

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I. Introduction

Probabilistic Safety Assessment (PSA) is an important tool to understand and further improve safety in nuclear power plants [1]. PSA methodology has been in continuous development over the years, driven by regulatory requirements and striving for precision, completeness, and accounting for uncertainties. This has led to very complex and detailed models that are difficult to use or understand outside the narrow circle of the developers or super-experts [2]. Despite their size and maturity, PSAs still suffer from some limitations [3], such as lack of completeness of the set of initiating events, lack of completeness and adequacy of causality models, limited treatment of organizational and human factors (operator error of commission, safety and organizational culture contribution, and others). Moreover, the current state-of-the-art PSA models are very detailed and plant-specific that aim -- and excel -- at supporting risk-informed decision making at plant level. Nevertheless, they are so specific that they cannot be used to understand industry-wide performance and big picture safety insights and trends.

For the sake of learning from international experienced events, gained over the long nuclear history, we are developing simplified PSA models for typical PWR and BWR designs. The models are not plant-specific and represent accident sequences at a generic level, focusing on lessons learned from analyzing hundreds of events from our curated database [4]. The mentality of going simple and generic – combined with empirical evidences – paved the way to account for important contributors of systems unreliability, which we fear are under-represented in regular PSAs.

Finally, the generic models will be used for order-of-magnitude precursor analysis of the operational events in our database [4]. The models will allow us to understand big picture safety issues, trends, and get better risk estimates by pooling worldwide experience from different plants and sites. We claim that our work can serve as a complementary framework to the existing PSAs, with more user-friendly models and different research objectives.

II. Methodology

A. Modeling Philosophy

Nuclear power plants vary in design and specific safety systems; nevertheless, they utilize the same general safety functions as defined by the IAEA and regulatory bodies [5, 6]. In our work, we are developing one generic set of event trees for PWRs and another for BWRs. Each set will span possible initiating events and accident-sequence scenarios, and will be coupled with generic fault trees to quantify event probabilities and the reliability of the safety systems and functions respectively. Plants will share the same set of event trees that include the same major systems/functions but might slightly differ in some features/configurations. Therefore, the event and fault trees will be customized to address important design-specificities, such as designed safety systems/functions and degree of redundancy.

Total loss of a support system (component cooling, offsite power) -- i.e. losses that affect the whole plant will be modeled as separate initiating events. Instrumentation and control functionality, will be decomposed into two parts, first, Instrumentation and measurement of physical variables (pressure, temperature, neutron flux, etc.) including data transmission, second, automatic actuation system (control algorithm, control signal transmission, etc.). The first part will be included in the event trees as an early top event that is supposed to be in-place, reliably measuring the required variables, so that any transient and irregular phenomena are detected[1]. The second part is included as a basic event in the fault trees representing the automatic safety system/function actuation signal.

B. Event Tree Modeling

In this section, we present one illustrative example of our event tree models currently under development. We describe a typical PWR response to a small break loss of coolant accident (SBLOCA) (see Fig. 1). In the event of a SBLOCA, the instrumentation and measurement system will detect the slowly decreasing pressure in the reactor coolant system (RCS) and issue a reactor and turbine trip signals. After shutdown, the reactor continues to produce heat (decay heat) requiring

[1] The instrumentation and measurement system consists of highly redundant and diversified detectors and logic circuits, therefore, its failure probability is very low. We decided to keep it for the time being as part of the event tree, however it could be screened out after further analysis if we realized its negligible contribution to the core damage sequences.

continuous cooling. Decay heat is removed through secondary-side cooling using the auxiliary feedwater (AFW) or the main feedwater (MFW) systems in coordination with steam dump system (turbine bypass valve and main condenser or the atmospheric relief valves). With the successful operation of the AFW or the MFW, the pressure of the primary coolant system is reduced to the point at which the high-pressure coolant injection system (HPI) can activate. Otherwise, if both feedwater systems are unavailable, a feed-and-bleed (F&B) operation can be performed to avoid core damage. Feed-and-bleed requires manual high-pressure coolant injection coupled with the opening of the power-operated relief valves (PORVs), manually or automatically depending on the design, to control the reactor pressure and discharge the decay heat to the containment. The RCS has to further depressurize until the residual heat removal (RHR) system – the low-pressure coolant recirculation system – can initiate. Finally, if the RHR system cannot operate because it is unavailable or because the RCS depressurization function failed, the high-pressure sump-recirculation mode can perform the long-term decay heat removal.

The presented event tree and the demonstrated safety systems serve as a skeleton model, which is customized – depending on the plant under analysis – to represent major design differences. Some plants might not have a feed-and-bleed function, and others can have a containment-spray recirculation system to support the long-term decay heat removal in case of RHR failure; these systems/functions differences can be easily incorporated as top events in the event trees. The resulting event trees are essentially similar to the event trees developed during the early days of the USNRC Accident Sequence Precursor Program [7].

C. Fault Tree Modeling

To quantify the failure probabilities of the top events, we are developing generic inclusive fault trees, which encompass human, organizational, and design aspects of failures in addition to the typical hardware and technical ones. Consider a general safety system with three redundant trains, i.e. 3×100%. The system/function can fail due to (see Fig. 2):

1) the failure of the three trains, independently or due to a common cause factor (CCF). CCFs cover all dependent failure causes that are not explicitly modeled in the fault tree.
2) actuation system failure (automatic or manual), including operator error of omission and the failure to manually recover a failed system, or
3) design residuals, include initial design issues and assumptions, incorrect systems actuation/trip logic, vendor/manufacturing error, etc…
4) organizational/regulatory and safety culture contributions, cost-cuts, risk communication, etc…
5) procedural errors which encompass both inadequate operator procedures and inadequate testing and maintenance procedures.

Currently, we hypothesize that design residuals, procedural errors, and safety culture are contributors at a system level, i.e. the trains are usually similar, and are likely to be affected mutually by these contributors. This claim could be modified or further confirmed in the course of analyzing more operational events.

To understand the causes of trains' unavailability, one can identify three possible reasons (see Fig. 3): technical, human, and testing/maintenance actions. The technical factor can be decomposed into two groups while having a train-level/super-component modeling philosophy:

1) frontline systems failures, which represents failures in major frontline components that affect the whole train functionality, i.e. suction valves, pumps, injection valves, etc…
2) "local" support systems failures, which affect a train within a specific safety system, it includes "local" component cooling system, "local" power supply and circuitry, local lubrication, etc.
3) "global" intersystem support failures, which include failures in support systems affecting trains in multiple safety systems.

The human factor could take one of two forms:

1) operator errors include errors of commission, tripping a functional train, rendering a system unavailable, etc.
2) maintenance crew errors cover errors committed during testing and maintenance actions, leftovers, wrong arrangements/adjustments, failure to detect apparent degraded conditions, inadequate repair, etc.

D. Basic Events Quantification

The current intention is to use our database [4] and pool statistics from worldwide experience to create estimates for the different contributors especially for the non-technical factors and important initiating events. When it comes to technical hardware failures (frontline and support), use of expert opinion might be sufficient because of the non-major contribution of these factors. If plant-specific analysts are not satisfied with these rough estimates, they can always zoom-in and use their plant-specific information and data for the respective parts of the fault tree.

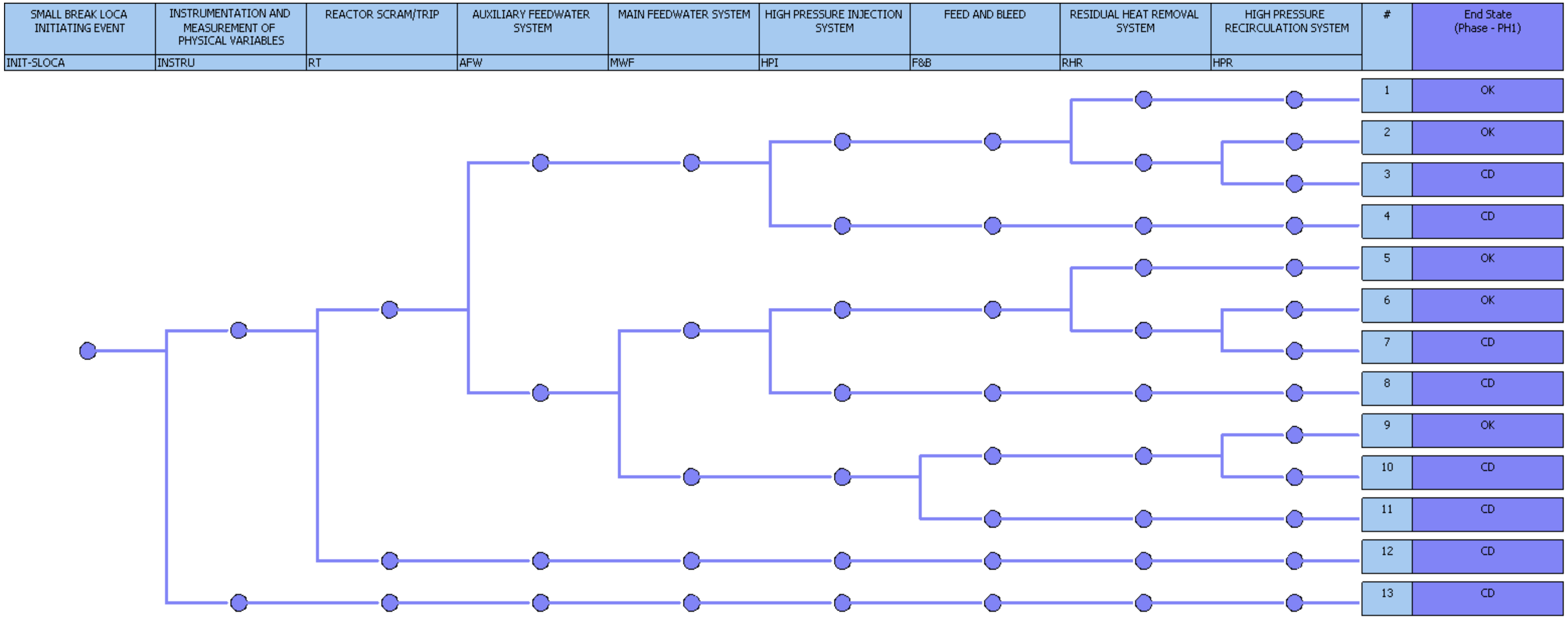

Figure 1. A generic PWR small break LOCA event tree.

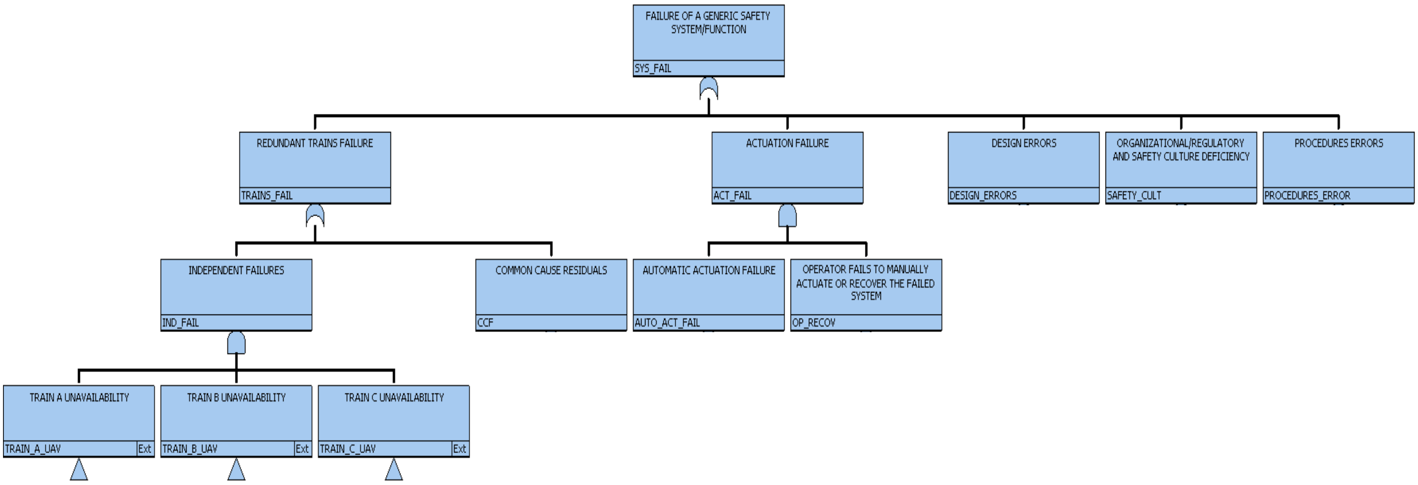

Figure 2. A generic system-level fault tree model.

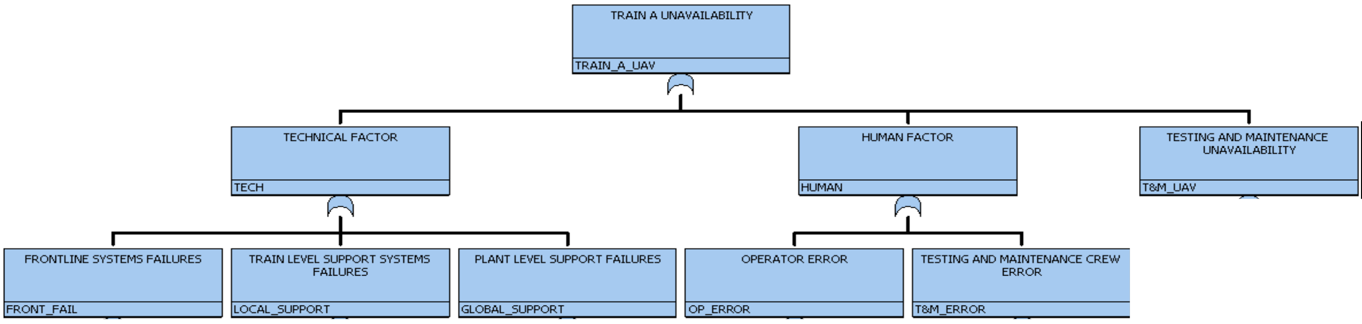

Figure 3. A generic train-level fault tree model.

III. CONCLUSIONS

In this work, we propose and present the first conceptual steps towards the construction of new generic PSA models that will be used for our international precursor analysis efforts. We are developing event trees and fault trees that are not plant-specific and represent general safety systems with a room for customization. The proposed models take into consideration lessons learned by analyzing nuclear operational events and mishaps, and incorporate empirically experienced contributors into the analysis supporting the quest for PSA completeness. We foresee them as a complementary framework in PSA analysis that can aid plant-specific PSAs and answer safety concerns on a different level.

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Systematic Approach to Transboundary Atmospheric Dispersion Assessment of a Hypothetical Release from a Nuclear Power Plant

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I. Introduction

It can be seen from past nuclear accidents at Unit 4 of the Chernobyl Nuclear Power Station (NPS) and Unit 1 – 4 of the Fukushima Daiichi NPS that the accidental release of radioactive materials from nuclear accidents are not just the problem of the country owning the power plant. The accidental release can migrate through the atmosphere or ocean to other countries [1, 2]. Therefore, even a country without a nuclear power program needs to be prepared for nuclear emergency in neighboring countries. ASEAN Network on Nuclear Power Safety Research (ASEAN NPSR) has been working on transboundary atmospheric dispersion assessment of a hypothetical release from adjacent nuclear power plants since 2017 [3]. Thailand, as a member of ASEAN NPSR, participated in the assessment using JRODOS, and found a possibility of detectable amount of radioactive materials reaching the boundary of Thailand under certain conditions [4]. This finding indicated the potential of using the national radiation monitoring system for preparedness and response to a nuclear emergency in other states.

Thailand Institute of Nuclear Technology (TINT), as Thailand's representative to ASEAN NPSR, jointly conducted a study with Chulalongkorn University (CU) and Office of Atoms for Peace (OAP) to reflect the results from transboundary atmospheric dispersion assessment to the central radiation monitoring system of Thailand [5]. It was found that a significantly large release (approximately 100 EBq of I-131 and 1 PBq of Cs-137) can trigger an alarm at specific radiation monitoring stations in Thailand when the wind blows toward them. However, thorough investigation of the conditions, e.g. NPSs of interest, potential release amount and timing, variation in meteorological data, variation in background radiation, has not been made. As the assessment integrate several different components of uncertainties, some of which are interrelated, a systematic approach is needed to accurately account for the effects from all relevant aspects. This study aims to propose a structured framework for a country without nuclear power plant to perform transboundary atmospheric dispersion assessment of a hypothetical accident from an external NPS, and effectively reflect the results to the radiation monitoring system in order to enable early detection and response to foreign nuclear emergency.

II. Proposed Framework

The proposed framework in Fig. 1 contains four steps. First, NPSs of interest are identified based on the distance from the boundary of the country. This study focuses on existing and planned power plants within 1,000 km from the boundary of Thailand, but the framework can also be used with virtual power plants or larger target area. The next step is determining the release characteristics. The best choice is to determine the release characteristics based on the accident scenarios in the safety analysis report (SAR) and the core inventory data of the NPS. Yet it is difficult to access this information, especially

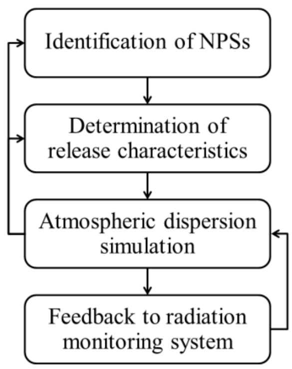

Figure 1. Proposed framework for transboundary atmospheric dispersion assessment.

when it is an external NPS. Release characteristics will most likely need to be determined based on publicly available data. In this case, it is important to confirm that the data used in the assessment is not detached from the actual conditions of the NPS, for example, the data should be taken from a similar type of reactor, and the accident scenarios should be typical ones for that specific reactor type. These conditions along with the meteorological conditions are then used to perform atmospheric dispersion simulation. Meteorological data must cover the target area, and the resolution must be sufficient. Each simulation code has its own advantages and drawbacks. Those employing Lagrangian particle model can represent the real world well and are appropriate for long-range dispersion (hundreds of km), though they require computational resources. By contrast, those employing Gaussian puff or plume model are more suitable for short-range (up to several km) and mid-range (tens of km) dispersion, but they can perform nearly real-time evaluation. This simulation outputs maps of air and ground concentrations of radionuclides and radiation dose. Simulation results can be fed back to upstream assessments, to consider different types of release, or to extend or shrink the target area which may increase or decrease the number of the source NPSs. The final step is to reflect the results of atmospheric dispersion evaluation to national radiation monitoring system. The level of artificial radiation dose rate detectable by each radiation monitoring station has to be determined based on the background dose and its annual fluctuation. This dose rate level is then compared with the simulated time-dependent dose rate of each accident scenario and each meteorological data set at the monitoring station. Based on the comparison, dispersion simulation of other possible accident scenarios or different meteorological data sets can be performed. Finally, this framework will output the accident scenarios and the meteorological conditions that are detectable by the radiation monitoring system, and the approximate time for its detection.

III. Example Assessment

A. Assessment Conditions

A 1,000 MWe unit of an anonymous NPS in China located approximately 600 km to the northeast from the boundary of Thailand is selected. The approximate location of the NPS concerning the boundary of Thailand is illustrated in Fig. 2. A radioactive release from a long-term station blackout scenario is used to perform atmospheric dispersion simulation. The release characteristics refer to publicly available data of the U.S. Nuclear Regulatory Commission [6]. Meteorological data is taken from National Oceanic and Atmospheric Administration (NOAA) website [7]. 24-hour weather data on November 24, 2018 at 11:00 PM (GMT+7), when strong northeastern wind and scattered rain can be observed, is used as an example. LASAT module of JRodos [8] is used for atmospheric dispersion calculation since it employs Lagrangian particle model. The calculation code outputs time-dependent hourly dose rate. Measured total dose rates from three national radiation monitoring stations in the northeastern region of Thailand recorded for a period of one year are statistically analyzed to determine the dose rate values to trigger the alarm. These are compared with the simulated results. Fig. 2 shows the locations of the monitoring stations and their relative positions and distances from the source NPS.

B. Example results

The time-dependent results of the maximum total dose rate at the boundary of Thailand and the total dose rate at three selected radiation monitoring stations of this specific case is shown in Fig. 3. The radioactive plume reaches the boundary of Thailand approximately 18 hours after the release, and reaches Station A an hour later. The plume reaches Station B 23 hours after the release, and does not reach Station C in the24-hour timeframe. Total dose rates at the boundary of Thailand and at Station A reach 1 nSv/h 20 and 21 hours after the release, respectively. For this specific case, dose rate at Station A can potentially be used to represent the dose rate at the boundary of Thailand.

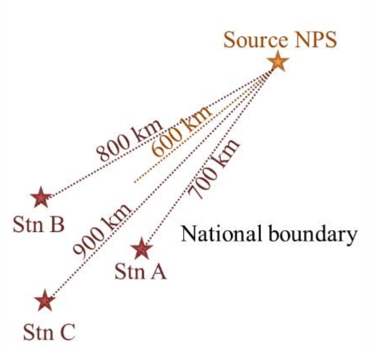

Figure 2. Simple illustration of the source NPS and the radiation monitoring stations.

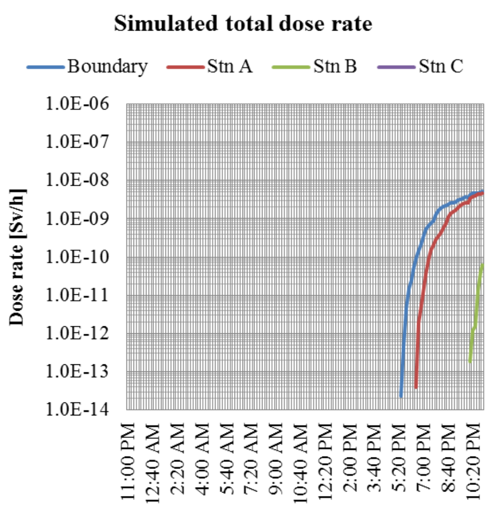

Figure 3. Total dose rate at the national boundary and monitoring stations simulated by JRodos.

TABLE I. STATISTICAL PARAMETERS OF MEASURED TOTAL DOSE RATES AT THE RADIATION MONITORING STATIONS

Statistical parameters	Dose rate [nSv/h]		
	Station A	*Station B*	*Station C*
Arithmetic mean	35.2	23.7	42.3
3-sigma	18.3	24.0	35.2

Average total dose rate at the radiation monitoring stations A, B and C is shown in Table I. These values represent the background dose rate at respective locations. Based on thorough consideration, the sum of the average total dose rate and the 3-sigma values is used as the alarm triggering value at the three monitoring stations [5]. This implies that the artificial radiation dose over the 3-sigma value would be detectable at respective stations. Comparing the values in Fig. 3 and Table I, the release from a long-term station blackout accident at 11:00 PM (GMT+7) of November 24, 2018 could not be detected by the monitoring system unless the release amount is larger by at least an order of magnitude.

IV. CONCLUSIONS AND STEPS FURTHER

A structured framework for a country without nuclear power plant to perform transboundary atmospheric dispersion assessment of a hypothetical accident from an external NPS was proposed. Four steps to be followed are identification of NPSs, determination of release characteristics, atmospheric dispersion simulation and feedback to radiation monitoring systems.

The example assessment indicated that a relatively large release from the source NPS during the period of the Northeast monsoon combining with scattered rain could potentially be detected by the national radiation monitoring system. Since the dose rate at the boundary and at one of the stations were in the same order of magnitude, it could be assumed that the current monitoring stations would be adequate for early detection of a large accidental release from an external NPS.

Note that these are insights from an example assessment of a specific accident in a specific NPS under specific weather conditions. The weather data is taken from a global domain with coarse resolution, and may not be able to capture the regional characteristics. Though the LASAT module seems to deliver promising results, it has not been validated for long-range dispersion calculation. In addition, statistical analysis of the radiation monitoring data covers only three selected monitoring stations, and only a single year of measured data.

Several tasks are to be done in order to complete the transboundary atmospheric dispersion assessment. ASEAN NPSR is performing an inter-comparison assessment with different calculation codes aiming to increase the reliability of the results. TINT, CU and OAP are working with Thai Meteorological Department (TMD) to prepare variety of weather data sets, some of which have higher resolution. TINT is also considering different accident scenarios including multi-unit and spent fuel pool accidents. In spite of the necessity to complete the remaining tasks, the results and findings from the example assessment suggest that the framework can help capture the big picture of the transboundary atmospheric dispersion assessment, and possibly facilitate the derivation of the findings which are useful for early detection and response to foreign nuclear emergency from the combination of analysis at different steps of the assessment.

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Utilization of Artificial Intelligence in the Analysis of Nuclear Power Plant Requirements

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I. Introduction

Nuclear power plant projects are often characterized by two factors: they are time-consuming and capital-intensive. These current challenges include descriptive and non-harmonized requirements demanded in the nuclear power industry resulting in the adaptation to a new licensing domain being very data-intensive, laborious, and tardy. Furthermore, the sheer volume of these requirements also poses a challenge [1], [2], [3]. Licensing and engineering could be facilitated and errors reduced in the allocation of requirements by utilizing artificial intelligence in the analysis of nuclear power plant requirements [4], [5], [6].

This study developed an algorithm capable of recognizing natural language to classify nuclear power plant requirements into predefined categories by utilizing supervised machine learning as well as examined atomizing requirements and addressing their similarities. The study was performed in close cooperation with an AI company, Selko Technologies Ltd, being responsible for the development of the algorithms based on the data and needs of Fortum Power and Heat Ltd (Fortum), a leading clean-energy company.

Fortum has developed a new high-level safety engineering method called *Advanced Licensing and Safety Engineering Method*, ADLAS®, to illuminate the licensing aspects and the safety features behind the licensing requirements [7]. The method is a systematic and well-documented way of preparing the licensing documents [8]. One of the long-term objectives is to implement an algorithm to perform according to the ADLAS hierarchy and methodology.

The classification algorithm involves a nuclear power industry-specific language model consisting of a recurrent neural network (RNN) with long short-term memory (LSTM) components, and a classifier based on a feedforward neural network (FFN). For training the classifier, a small selection of the Finnish regulatory requirements (YVL Guides) issued by the Finnish Radiation and Nuclear Safety Authority (STUK) [9] were classified according to the two-level predefined requirements hierarchy. The algorithm was tested on the selected YVL Guides and a set of requirements issued by the Office for Nuclear Regulation in United Kingdom [10].

II. Theory

A. Artificial Intelligence and Deep Learning

Artificial intelligence can support reaching engineering goals in helping to decide and exclude certain logical tasks performed by an expert. The challenge is to solve tasks which people can easily perform but hardly formally describe [11], [12].

Deep learning has been enhanced ever since the AlexNet [13] secured first place in ImageNet challenge; furthermore with the use of GPUs to deal with the computationally intensive operations of deep neural networks (DNNs). DNNs in essence comprise of convolutional neural networks (CNNs), recurrent neural networks (RNNs) and feedforward neural networks (FNNs). The basic idea of these networks comes from the idea of multi-layer perceptron (MLP), which can learn complex non-linear functions. The MLP is a multilayer network, a mathematical function composed of many simpler functions, mapping input values to output values [12], [14].

B. Natural Language Processing

Human language is complex and for a machine to understand the meaning (i.e., structure of language), we make use of natural language process (NLP), a study of language using linguistics, computer science and machine learning. Deep learning for NLP uses RNNs to model sentences and modelling long sequences face a lot of major challenges, since it requires to keep track of entire sentence structure. Long short-term memory networks were utilized in order to memorize the large sentences. A LSTM unit consists of several other neural networks, having individual tasks, such as ignoring, forgetting and selecting words. The idea of LSTM is being capable of remembering the events occurred since many time steps [15]. The architecture of RNN is applied in the LSTM models as a short-term memory of RNN is combined with a long-term memory of LSTM [16].

III. ALGORITHM DEVELOPMENT

A. Classification of Requirements

The requirements classifier was developed for categorizing requirements according to two-level hierarchy. In practice, there are two classifiers; the first one classifies requirements into four upper-level requirements (Heading, Process, Technical and Reference). The second classifier further categorizes only process requirements into five lower-level categories (Design, V&V, Qualification, Documentation and Licensing).

To classify requirements, the developed natural language processing (NLP) algorithm consists of two major components; a language model and classifier; utilizing and modifying the model from [17]. The language model is based on recurrent neural network involving long short-term memory networks. The architecture of the language model consists of three layers of LSTMs followed by maximum, average and minimum pooling methods. Finally, concat pooling is used to form a feature vector representing a requirement in the form of vector. The model learns the language by predicting the next word in a sentence. The network can be trained to remember things from previous words when an array of words is sequentially processed. The language model was trained on Wikipedia dataset [18] followed by three regulatory datasets issued by STUK [9].

In the training phase of the classifier, predictions were compared with the ground truths issued by a human expert in the form of classified training data. To minimize the error between prediction and ground truth, the results were backpropagated to the beginning of FFN. Backpropagation through time for text classification (BPT3C) was utilized resulting in a separate computation of the cost of each time step. Consequently, the weights of all networks (e.g., the RNN with LSTMs, FFN, and networks inside each LSTM cell) are concurrently adjusted for the classification task based on the backpropagation. The only controllable element is the loss function which is binary cross-entropy in this case. After the classification task, we continued developing methods to atomize requirements and indicate the similarity of them.

B. Atomizing Requirements

Long and ambiguous requirements should be atomized to facilitate the allocation of requirements for each design discipline. The challenge is to divide a multi-class requirement to result a single-class demand which would facilitate the allocation of requirements.

In the context of the paper, we define "atomization" as a process of breaking down complex requirements into simple and coherent sentences, which still keep track of requirements context. A simple pipeline would involve splitting sentences with determinants, even such a simple approach resulted in better accuracies. We further used Stanford parser [19] for NLP parsers and Coreference resolution [20], to further improve the atomization pipeline.

A single YVL Guide was used to test the model. The model performed well according to the set rules but the challenge was to preserve the context of a requirement when one is separated into two or more clauses. However, the same challenge occurs when an expert atomizes manually a long requirement. In addition, there are cases in which an atomized sentence lacks an essential end of the initial sentence.

C. Similarity of Requirements

When a requirements specification include various national and international requirement sources, it is very likely that the specification contains many similar demands. Therefore, it should be able to combine them to reduce the size of the requirements specification and thus, not every native requirement is required to be further allocated.

Expanding on the ideas of attention mechanism in [21] and transfer learning from ELMo [22], OpenAI GPT [23] introduced Transformer which encode the input text using a series of multi-head attentions. The text representation from the transformer encoders started outperforming, empirically, the previous state-of-the-art benchmarks. Recently, BERT, short for Bidirectional Encoder Representations from Transformers [24], showed impressive results in all downstream NLP tasks such as text classification and sentence tagging. We used the BERT embeddings generated from textual requirements for sentence similarity task in which we compute the L2 distances of two sentence embeddings. We say two requirements are similar if the L2 distance is close to zero. It is also important to understand that achieving good semantic similarity is a hard problem in NLP, as a slight change in structure of the text can completely change the meaning of the sentences.

IV. EXPERIMENTAL RESULTS

Accuracies of the classification model and the corresponding thresholds are tabulated in Table I. The accuracies were calculated by first computing the fraction of incorrectly predicted labels to the total number of labels, that is, Hamming Loss (HL).

TABLE I. MODEL ACCURACY AND THE CORRESPONDING THRESHOLD IN EACH CLASSIFICATION TEST

Test Dataset	Threshold	HL	Accuracy
1/3 of Initial Data (upper-level classes)	0.90	0.10	0.90 (90 %)
1/3 of Initial Data (incl. subcategories)	0.90	0.20	0.80 (80 %)
Blind YVL B.2 (incl. subcategories)	0.70	0.14	0.86 (86 %)
Blind UK SAP (incl. subcategories)	0.50	0.15	0.85 (85 %)

Table II presents samples of the correct classifications. In the table, probabilities show model's degree of certainty at which a requirement belongs to a certain category. In this case, the predicted labels equal to the ground truth.

TABLE II. SAMPLES OF CORRECT CLASSIFICATIONS

CI ID	Description	Probabilities
YVL-E.7-3.4.2-351	351. In connection with the final suitability analysis of electrical or I&C equipment in safety class 2, an independent assessment of the acceptability of the qualification procedure shall be presented.	Process = 1.00 Qualification = 1.00 Documentation = 1.00
YVL-B.1-5.4.2-5425	5425. The plant unit's power supply systems shall be dimensioned to supply sufficient electrical power for the implementation of the safety functions in all plant conditions.	Technical = 1.00

Table III presents two original requirements each of them being split into two individual sentences separated by the conjunction "and". The first one has been correctly atomized but in case of the second requirement, the latter atomized sentence should also end in "licence application". The second sample highlights one of the prime concerns when atomizing requirements; they should retain the semantic meaning and context of the initial requirement.

TABLE III. SAMPLES OF ATOMIZED REQUIREMENTS

Original Requirement	Atomized Requirement
During the application for a construction licence, STUK assesses the appropriateness of the preliminary system-level safety classification document required under Section 35 of the Nuclear Energy Decree (161/1988) and approves the preliminary classification document.	During the application for a construction licence, STUK approves the preliminary classification document.
	During the application for a construction licence, STUK assesses the appropriateness of the preliminary system-level safety classification document required under Section 35 of the Nuclear Energy Decree (161/1988).
Sections 35 and 36 of the Nuclear Energy Decree contain the requirement that the classification document be submitted to STUK as part of the construction and operating licence application.	Sections 35 and 36 of the Nuclear Energy Decree contain the requirement that the classification document be submitted to STUK as part of operating licence application.
	Sections 35 and 36 of the Nuclear Energy Decree contain the requirement that the classification document be submitted to STUK as part of the construction.

Table IV presents samples of detecting similarities between requirements of two comparable sources set by STUK [9], [25]. The similarity is annotated with a score. The smaller the score, the more similar two sentences are semantically. The results indicate that similarities can be discovered between two requirements based on similar words and word orders.

TABLE IV. SAMPLES OF SIMILARITY DETECTION

STUK Y/1/2018	YVL Guide B.1 Chapter 4	Score
Systems, structures and components important to the safety of a nuclear facility shall be available as detailed in the design basis requirements.	411. If shared structures, systems and components important to safety are designed for nuclear power plant units located on the same plant site, it shall be demonstrated by means of reliability assessments that this does not impair the capability of these structures, systems and components to perform their safety functions.	4,616
A nuclear power plant shall have the necessary components and procedures for securing the removal of residual heat from the nuclear fuel in the reactor for a period of three days independently of the off-site supply of electricity and water in a situation caused by a rare external event or a disruption in the on-site electrical distribution system.	452. The nuclear power plant shall have in place arrangements that can guarantee sufficient cooling for the fuel placed in fuel storage facilities during rare external events in accordance with requirement 450. These arrangements shall make it possible to supervise the water level in the spent fuel pools for a minimum of eight hours without recharging the DC batteries. Furthermore, it shall be possible to keep the fuel reliably submerged during the loss of the plant's internal electricity distribution system in accordance with requirement 451. A sufficient inventory of water and fuel and capability to recharge the DC batteries shall exist at the plant site to maintain these arrangements for a period of 72 hours.	4,636

V. DISCUSSION

The main challenge in utilizing narrow AI in requirements analysis is that the current parameters (i.e., weights and biases) cannot be employed in another case, such as classifying requirements into other categories than those utilized in the training dataset.

The classifier had difficulties in categorizing requirements which contained unfamiliar terms or word orders compared to the samples in the training dataset. This challenge was emphasized when the model was tested on two blind datasets which discuss slightly different issues within the same domain. Furthermore, specific terms and their context vary within various regulatory domains.

The future classifier has to be trained on data structured according to ADLAS hierarchy including various levels such as plant level, architecture level, system level and equipment level in order to facilitate engineering and licensing processes [8]. However, allocating native requirements is not sufficient to improve the processes but requirements should be elaborated and further allocated to individual lower-level disciplines. This method would improve transparency.

The requirements analysis is generally a challenging process because of ambiguous statements. The challenge is not a country-specific because it involves internationally all parties, that is, a regulating authority, supplier, and licensee or client. We propose that further investigations should focus on studying the following issues listed below.

- Atomizing complex requirements
- Combining similar requirements
- Checking requirements syntax
- Meta learning

A stakeholder requirement is typically long and may require many demands even for various disciplines. Multi-class requirements should be atomized to result in single-class demands while preserving the context of a parent requirement. Thereafter, similar requirements should be combined to reduce the amount of requirements in a requirements specification. EARS syntax should always be checked to enable coherent requirements. Finally, meta learning could be experimented to form appropriate training datasets with only a few training samples together with a description of the content of each category.

VI. CONCLUSIONS

As the results of this study are promising, the research is recommended to be continued. There are several practical proposals for the future studies which would facilitate requirements engineering processes in nuclear power industry and thus, provide a vast amount of cost savings used in the complex licensing and engineering processes.

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Measurement of scintillation light yield in NaI(Tl) detector for the SABRE dark matter experiment

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I. INTRODUCTION

Scintillation detectors have numerous applications in nuclear science, experimental particle physics, medical physics, environmental studies and many other areas [1], [2]. The high light output, optimum energy resolution, standard light yield proportionality and lower production costs are the prerequisite for the scintillation materials to make a good scintillation detector [1], [3], [4], [5], [6]. The development of inorganic and organic scintillation detectors has been tried over the past few decades due to different industrial and research applications [6], [7], [8], [9]. Plastic scintillators are used for neutrons and charged particles detection and spectroscopic applications although its light output is lower than most of the inorganic scintillation detectors due to lower fabrication cost, fast decay time of light pulse and easy to use. In 1948 NaI(Tl) inorganic scintillation crystal was discovered and it is being widely used and most known scintillation crystal to make scintillation detectors with high light output and energy resolution [7], [8], [9], [10]. To understand the scintillation process in NaI(Tl) scintillation detector over the past few decades, significant amount of research has been carried out across the world; however, many aspects of NaI(Tl) detector are not fully understood to date [7], [8], [9]. A major complication in NaI(Tl) detector is that the proportionality between the incident particle energy and the number of photons generated , known as the scintillation light yields, is not always constant. In NaI(Tl) detector there is a well-known non-linearity for gamma ray photons and electrons production depending on energy deposition in scintillation crystal ,which could affect the energy resolution and scintillation efficiency of NaI(Tl) detector. In the case of heavy charged particles recoiling in Na (sodium) or I (Iodine) nuclei of scintillation crystal, the scintillation light yields are only a small fraction of that for electrons or photons productions inside the scintillation crystal [1]. In dark matter detection experiment, dark matter signals result from the recoil of either sodium or iodine nuclei in the scintillation crystal [4]. SABRE (Sodium-Iodide with Active Background Rejection) dark matter experiment based on NaI(Tl) scintillation detectors will be operated in southern (Staywell Underground Physics Laboratory, Melbourne, Australia) and northern hemisphere (INFN, Italy) simultaneously for 3- 5 years to verify the DAMA/LIBRA dark matter detection claim [4]. Therefore, it is imperative to know the light yield non-proportionality in NaI(Tl) crystal and quenching factor of Na and I nuclei recoils. Light yield non-proportionality can be measured in NaI(Tl) detector as a function of either electron response or photon response [8]. Electron response is the ratio of light yield to the energy of electron and it is an intrinsic characteristic of scintillating material like NaI(Tl). Porter et al. have studied the electron response nonproportionality of different scintillation detectors with external electron source, which could be affected by surface effects in scintillation materials [8]. Rooney et al. and Valentine et al. have introduced the Compton coincidence technique (CCT) to measure the light yield nonproportionality in NaI(Tl) detector [7]. The Compton coincidence technique can measure the electron response in the range 3- 450 keV more accurately because the monoenergetic electrons are produced inside the scintillation crystal by γ ray energy deposition in the NaI(Tl) scintillator [7], [8].

The main aim of this study was to investigate the light yield non-proportionality in NaI(Tl) detector as a function of electron response for SABRE dark matter experiment.

II. EXPERIMENTAL METHOD

In the conduced CCT experiment shown in Fig. 1, a 662 keV ^{137}Cs (10 μCi) radioisotope was used as γ-rays energy source with close proximity to NaI(Tl) detector to measure the electron response in the scintillation material of NaI(Tl) detector as a function of electron energy. The position of ^{137}Cs gamma source close to the NaI(Tl) detector was varied from 30^0 to 135^0 for understanding the Compton scattered gamma photon energy deposition in HPGe (High Purity Germanium) detector. The NaI(Tl) (905-3, 2 x 2 inches crystal, 2 inches tube) and HPGe detectors were purchased from ORTEC, USA for this study. All used detectors and gamma source for CCT experimental set up were mounted on a circular stage to facilitate the position change of ^{137}Cs radioisotope. In the conducted CCT experiment, 12 different scattering angles were selected to collect the HPGe (Compton scattered energy) and NaI(Tl) (recoil electron energy) spectra, which was finally used to calculate the scintillation light yield non-proportionality. The schematic of the nuclear instrumentation used in this study is shown in Fig. 1. Each gated NaI(Tl) energy spectrum was collected for around 12 hours to allow sufficient counts and each gated HPGe spectrum was collected for 6 hours.

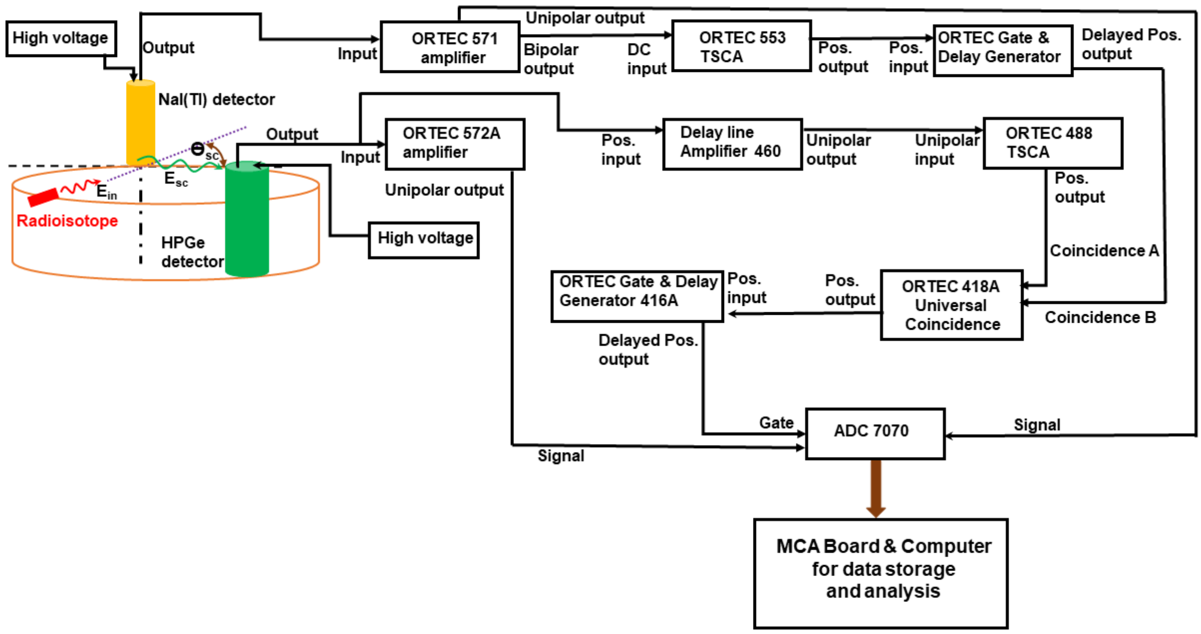

Figure 1. Schematic diagram of Compton coincidence technique (CCT) with electronics setup to measure electron response in NaI(Tl) detector as a function of electron energy.

The energy of scattered photon (E_{sc}) can be calculated by [8],

$$E_{sc} = \frac{E_{in}}{\left(1+\frac{E_{in}}{m_0c^2}\right)(1-cos\theta_{sc})} \qquad (1)$$

In the Fig. 1 and Eq. 1, E_{in} is the incident photon energy, E_{sc} is scattered photon energy, electron rest mass energy (m_0c^2) is equal to 511 keV, ϕ is electron recoil angle and θ_{sc} is the Compton scattering angle. In fact, the probability of Compton scattering at an angle can be estimated from the Klein-Nishina equation [4].

III. RESULTS AND DISCUSSION

Fig. 2 shows the energy spectrum of ^{137}Cs radioisotope source which was used to calibrate the NaI (Tl) detector, when the energy resolution was around 9.5%. The energy calibration was also done for HPGe detector. To understand the relationship between channel number and energy (keV) for both of the detectors, energy calibrations were conducted using ^{137}Cs and ^{133}Ba radioisotopes. Energy spectra from NaI(Tl) detector based on CCT were fitted with user defined Gaussian-Step function and ORTEC MCA software, and then mean recoil electron energy was calculated using detector's calibration data. In addition, the energy spectra from HPGe detector was fitted with Gaussian function only. The energy response for both of the detectors in CCT at Compton scattering angle 30^0 is shown in Fig. 3. The recoil electron energy in NaI(Tl) detector is increasing with increasing the Compton scattering angle shown in Fig. 4, which is opposite to the scattered energy deposition in HPGe detector. The obtained electron recoil energy (E_e) from NaI(Tl) detector with FWHM according to MCA software fit and Compton scattering angle is given in Table 1. The recoil electron energy deposition in NaI(Tl) detector at scattering angle 90, 100 and 110^0 shows relative inconsistency with overall energy deposition trend due to detector's less energy resolution and photomultiplier noise effects, because the NaI(Tl) detector's energy resolution usually decreases with increasing the electron energy in CCT and the photomultiplier signal can be affected by heating effects [1], [7], [8], [9]. The total energy response of NaI(Tl) detector in CCT depends on geometric component and electron energy resolution subtended by the HPGe detector. The total energy response from NaI(Tl) and HPGe detectors in CCT was not equal to the 662 keV ^{137}CS γ-rays value always, which might be due to the detectors geometric component and data acquisition system [8].

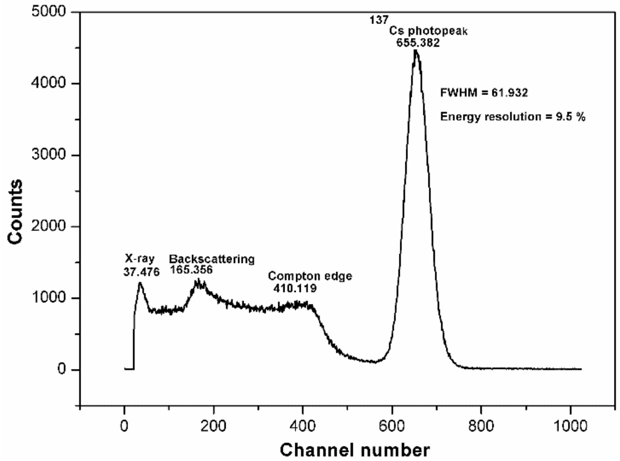

Figure 2. Energy spectrum of 662 keV γ- rays from ^{137}Cs radioisotope source measured with NaI(Tl) detector for calibration purpose.

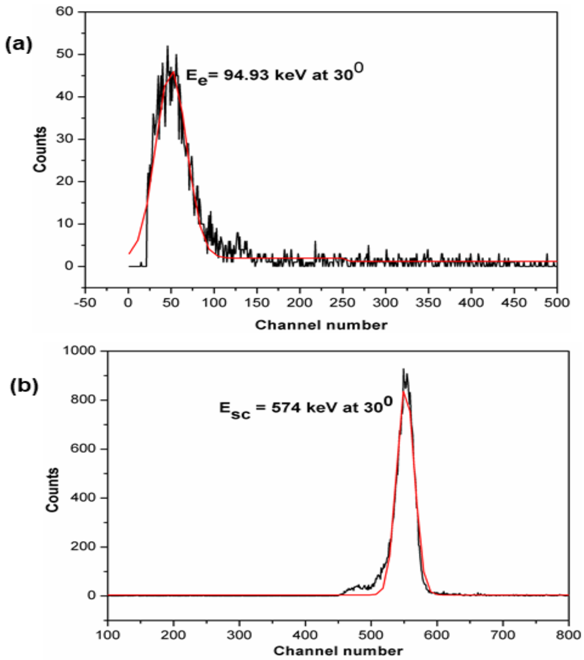

Figure 3. The energy spectra at 30^0 scattering angle in CCT with fitting: (a) recoil electron energy peak from NaI(Tl) detector, and (b) Compton scatted energy peak from HPGe detector.

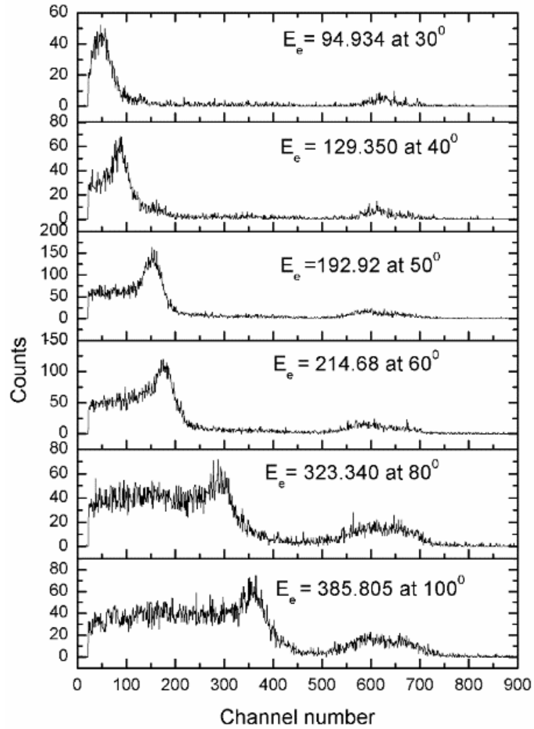

Figure 4. Energy spectra from NaI(Tl) detector in CCT at different scattering angle.

TABLE 1. MEAN RECOIL ELECTRON ENERGY FROM MCA SOFTWARE FIT WITH FWHM AT DIFFERENT SCATTERING ANGLE.

Electron recoil energy (E_e) in NaI(Tl) in keV	FWHM in channel number	Compton scattering angle (Degree)
94.934	24.72±1.4	30
129.350	26.540±0.91	40
190.92	34.734±0.62	50
214.68	36.47±1.1	60
286.09	35.65±1.3	70
323.340	37.677±1.1	80
294.626	33.79±1.4	90
385.805	33.350±1.6	100
369.85	45.674±1.4	110
406.27	46.366±1.5	120
415.321	48.86±1.7	130
423.60	54.16± 2.5	135

The total energy response in CCT can also be affected with HPGe detector's energy calibration data, which could result in incorrect Compton scattered energy deposition value [10]. In addition, multiple Compton scattering in scintillator can also increase the systematic error for measuring the total amount of energy deposited in scintillator based on CCT. The electron response or relative light yield in NaI(Tl) detector using CCT measurement was calculated as the ratio of detected electron recoil energy ($E_{NaI(Tl)}$) in the NaI (Tl) detector and the actual recoil electron energy (662 keV - E_{HPGe}) . The electron response,

$$R = \frac{E_{NaI(Tl)}}{662\ keV - E_{HPGe}} \qquad (2)$$

The R was plotted in Y- axis as a function of the actual recoil electron energy (X- axis) shown in Fig. 5. All the four curves for electron response as a function of electron energy was normalized at 440 keV to compare with each other precisely.

Calculated electron response based on conducted CCT experiment shown in Fig. 5 is well agreed with previously published results, which confirms that the CCT is providing the accurate measurement for relative light yield non-proportionality of NaI(Tl) detector within the experimental uncertainty [2], [7, [8], [9]. In Fig. 5, the calculated results indicate a nearly proportional electron response above 150 keV (3.5% change). Below 150 keV, the electron response increases more significantly, indicating a higher light yield nonproportionality for the low electron energy.

The higher light yield nonproportionality at low electron energy in NaI(Tl) scintillator could be due to the non-uniform ionization density, photomultiplier thermal noise, scintillator size and crystal quality. With the state-of-the-art electronics and detectors, more accurate measurement of light yield non-proportionality of ultrapure NaI(Tl) crystal in CCT could be achieved in the electron energy range 2 – 450 keV. Another CCT based light yield nonproportionality measurement in ultrapure NaI(Tl) scintillator will be carried out at 5^0, 10^0 and 15^0 Compton scattering angles with higher energy γ-rays source for understanding the electron response at low electron energy. In conclusion, the light yield nonproportionality data obtained will be useful for the SABRE dark matter experiment.

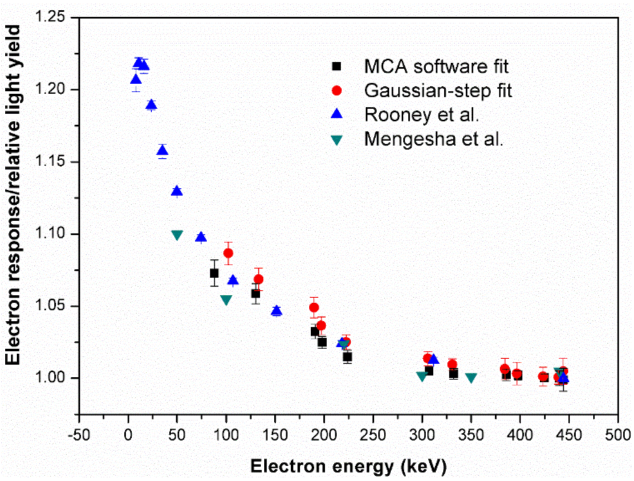

Figure 5. Relative light yield response in NaI(Tl) detector versus electron energy using CCT normalized at 440 keV arbitrarily.

ACKNOWLEDGEMENT

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Decommissioning a fuel assembly manufacturing plant in Belgium: health physics aspects from first methodology to final release

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I. Introduction

FBFC International is a former fuel assembly manufacturing plant currently under decommissioning in Dessel, Belgium. Founded in the early 1960s, it produced uranium fuel assemblies of light enrichment starting from uranium oxide powders, and later on also bundled MOX fuel rods into assemblies. Production ceased in 2012, respectively 2015 and the decision was taken to completely dismantle and denuclearize the plant.

This paper aims at providing the reader with an overview of the clearance methodology for both buildings and terrain, as well as illustrating those principles with some figures and practices that might serve as a basis for future decommissioning projects.

II. Clearance of Buildings

A. Clearance methodology

The first step is defining how one will demonstrate the absence of residual radioactivity in order for the building to be cleared from regulatory control and be demolished. This methodology is subject to approval by the nuclear regulator and their technical support organization (TSO).

The residual radioactivity is present in the form of a superficial contamination with uranium-oxide as production did not take place in glove boxes. There is no in-depth activity present as might be the case with activation in particle accelerator bunkers or in the high neutron flux environments near reactor pressure vessels. Dealing with fresh fuel, the isotopic vector is fairly simple and consists only of U-234, U-235 and U-238. Their ratios could be determined by the average enrichment processed over the lifespan of the respective buildings and lay between 2%-4%.

Surface contamination measurements were selected as the principal method to prove the absence of any residual activity. Every room in the building was assigned a risk category based on former use, each requiring a different level of control. In former controlled areas with risk of contamination, 100% of the surface (floor, walls and ceiling) was measured. For adjacent rooms and controlled areas without risk of contamination, this could be relaxed to 10%. Finally, certain rooms such as offices, canteen, … were excluded from measurements.

The surface is considered free from contamination if the measured surface activity is below 0.04Bq/cm² (total alpha) and 0.4Bq/cm² (total beta-gamma). These limits are widely in use, e.g. in transport regulation for dangerous goods [1], and had already been used on site before, still during production, when decommissioning a small disused building. It is worth noting that other standards exist that quote nuclide specific limits, such as RP113 [2], that could be used as a basis as well.

Any surface exceeding these limits is decontaminated based on mechanical abrasion: shaving with an integrated vacuum cleaner that diverts the dust immediately towards a 220 liter waste drum or using a jackhammer if more material needs to be removed. The surface can then be remeasured.

Once a room is deemed free of residual contamination, a second measurement is performed by personnel of the health physics department, albeit at a lower percentage as a means of verification.

The compendium of measurement results is presented to the regulator whose TSO also performs independent verification measurements of their own. Only afterwards, the licensee obtains permission to demolish the building.

B. *In practice*

In practice, there was around 35000m² of building surfaces to verify. The measurements are mainly performed by subcontractors who are trained at FBFC International by the health physics department. It consists of a theoretical part with evaluation, followed by hands-on training and certification. Different levels of qualification exist depending on the difficulty level: large flat surfaces, rugged surfaces, surfaces smaller than the instrument.

Two types of surface contamination monitors are in use, the VF Radhound SFP-100C and Berthold LB124SC300 with active surfaces of 100cm², respectively 345cm². The actual choice depends on the surface to be measured, but remains well below the maximum allowed surface of 1m² [3].

Each contamination monitor is designated a unique operational limit precalculated by taking into account device efficiency, maximum allowed background rate and a preset integrated measurement time of 10s before moving the detector to the next position.

Surfaces are split up in square regions of 1m² with a unique identifier and operators note down the highest value per m² on a form for central registration and compare them to their operational limit to decide if decontamination is required or the surface can be free released.

Practical measurement rates are around 8m²/(operator.day), respectively 16m²/(operator.day) depending on the contamination monitor used.

III. Soil Remediation and Clearance

A. *On-site*

Not only the absence of residual contamination in buildings has to be demonstrated in order for the clearance of FBFC International from regulatory control, but also its premises.

Site survey is based on the EURSSEM [3]. Much like the buildings' surfaces, the site is split up into areas of different risk categories based on their history or the one of the buildings erected there:

- Class I: expected contamination above the free release level (1Bq/g);
- Class II: no contamination above the free release level expected, increased activity levels possible;
- Class III: very low risk of contamination.

For each class, a different number of sample positions is required to reject the null hypothesis that the area is contaminated with sufficient confidence. The exact number depends on the expected variation of measured activity concentration in such a class. This can be derived from a preliminary site survey. At FBFC International however, results from a prior remediation project of more limited scope in 2001 were reused to this end. The positions are then distributed over the area following a periodic grid for class I and II, or put randomly for class III.

It is worth pointing out that EURSSEM recommends an additional 100% screening of the ground for hotspots for class I, e.g. using a mobile gamma spectrometer. However, given the radionuclides in play at FBFC International (U-234, U-235 and U-238 only), the most significant gamma emission line being at 185.7keV (U-235) this would result in a screening depth too little to serve its purpose. Instead, it was opted to place additional sampling positions near all potential sources of hotspots: along piping pathways, around and under retention pits and along foundations.

The sampling positions are then laid out physically by a surveyor. For each position soil samples are taken at 50cm depth interval down to 2m by default. As experience with preliminary surveys showed in case of soil contamination, the activity concentration goes down with increase in depth, the sampling series could be terminated earlier in case nothing was found in the first three samples. It is worth pointing out that this approach should not be generalized without thought, as local soil stratification or different radionuclide inventory might favor a different activity profile in depth. Moreover, in case of extra sampling along underground piping pathways, the starting point of the series should not be chosen at surface level, but at piping level (-1m).

Samples are mainly retrieved manually using a ground drill. About 300g of soil is retrieved at each required depth, corresponding to a depth averaging around 10cm. They are dried and spread evenly on a fixed reference surface for alpha surface activity measurement, an empirical factor subsequently being used to convert it to activity concentration levels. Though less precise than gamma spectrometric analyses, this custom method allows for fast treatment of large numbers of samples: up to 90 measurements/(operator.day).

The volume to be excavated is then defined as the bounding volume defined by sample points < 1Bq/g around the contamination, but also taking into account practical concerns as grouping of small neighboring volumes and having an excavation slope < 45° to avoid the created pit from caving in.

Sample positions are marked out again by the surveyor to verify all contamination has been removed and to map the excavated surface for record keeping.

The excavated soil is buffered in a storage tent and then guided through a dedicated system for radiological measurement and sorting, designed and operated by NTES [5]. The soil is forced into a fixed geometry by conveyor belts and run past two high-purity germanium spectrometers that pick up the 185.7keV line for U-235 to determine the activity content. Together with a scale mounted on the same belt, the activity concentration of the soil can be determined per batch of ±100kg at speeds around 10 ton/hour. The batch of soil is then directed towards one of three possible output streams:

- < 1Bq/g: free release for any use. In first instance, this sand will be used to refill the excavation pits;
- 1-10Bq/g: conditional release. With a license according to article 18 of the Royal Decree of 20/07/2001, this material is transferred in big bags to a conventional landfill;
- ≥10Bq/g: transferred to the national radioactive waste management agency (ONDRAF/NIRAS).

B. *Off-site (public domain)*

Limited uranium deposition was also discovered in the ditches, outside the site's perimeter, samples indicating light activity concentrations limited to 6 Bq/g. During the production period, after treatment of process waters and screening of both environmental and radiological parameters, these waters were released in the ditch running past the company. However, a lowering of free release levels and gradual sedimentation over more than 50 years of production, have led to the ditch soil exceeding the current limit of 1Bq/g. The decision was subsequently taken to remediate them over a total cumulated length of about 2km. Community meetings were held to inform the public.

A dose assessment ensured the dose limit for the public of 1mSv/y would not be exceeded during remediation, greatly facilitating the works, not having to consider subcontractors professionally exposed workers. Nevertheless, some basic principles in preventing contamination spread were applied:

- Clothing proper to the excavation site;
- Container for changing and showering;
- Equipment, trucks, bulldozers, dedicated to the site for the entire duration of the project and measured for free release on demobilization.

In total, 24000 tons of soil have been excavated and run through the sorting equipment of NTES. Of this soil, 23000 tons (representing 96%) were free releasable. An argument could thus be made for higher activity concentration limits for soil screening, not unlike existing for buildings [2] and metals [5], instead of current general ones for free release.

IV. Conclusion

The decontamination and decommissioning of FBFC International is well under way, with only the measurement and demolition of one nuclear building left to perform, followed by the site remediation.

Free release methodologies for buildings and terrain have been introduced and might serve as a guidance to similar projects in the future.

As a final step before the decommissioning license is lifted and the site officially cleared from regulatory control, a final decommissioning report will have to be drafted and approved by the authorities. It provides an overview of all dismantling activities and radiological surveys performed.

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Export Controls in the Supply Chain of Nuclear Products and Technologies. Theory and Practice

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I. Introduction

Export Controls (EC) are a set of measures to ensure the implementation of foreign economic activities (FEA) in respect of goods, information, works, services, results of intellectual activity which can be used to create weapons of mass destruction (WMD), means of its delivery, other types of weapons and military equipment or when preparing or committing terrorist attacks. Thus, export control procedures make a significant contribution to effective compliance of states and enterprises with international non-proliferation treaties and are aimed at minimizing the risks of incidents in terms foreign trade activity, which relates to dual-use products and can undermine international peace and security. This paper is devoted not only to the issue of the implementation of export controls of dual-use goods of certain types in the context of law enforcement and to the economic and political aspects of export control but also to the overview of the system at different levels (international, national and corporate) and presentation of the best Russian practice.

II. Export control for nuclear transfers

The implementation of export control procedures when conducting transactions with foreign counterparties regarding the nuclear goods and technologies is the most important condition for the nuclear non-proliferation. The relevance of this paper is justified by labor costs optimization and risks minimization of the foreign economic activity carried out by Rosatom enterprises.

III. Rusatom Service export control compliance

A. *Rusatom Service Portfolio*

Rusatom Service JSC (Joint Stock Company) was established in 2011 as a wholly owned subsidiary of Rosatom State Corporation. Rusatom Service is a member of Electric Power Division of Rosatom and plays a role of service offer integrator of Rosatom State Corporation abroad. The company provides full range of services and supplies required for maintenance and repair for the foreign nuclear power plants that operate VVER-type reactors. Rusatom Service holds leading positions on the markets of China, Bulgaria and Armenia and acts as a general contractor of works for lifetime extension, implementation of scheduled preventive maintenance and upgrading the equipment at VVER nuclear power plants.

B. *EC System: Corporate Level (Basic)*

Rusatom Service is a guarantor for the implementation of the licensing procedure for operations, compliance with international and national requirements for export control. Rusatom Service has its own Internal Compliance Program which allows to provide different types of operational and business activities such as obtaining permits, personnel training on EC basics both for its own and contractor’s personnel. Besides, provision of EC services (contract management, comprehensive consulting services, preparation of a set of documents).

IV. Assurances while exporting to NSG (Nuclear Suppliers Group) members and non-NSG members

A. *Case Study*

Nuclear Suppliers Group (1974) is a multilateral export control regime and a group of nuclear supplier countries that seek to prevent nuclear proliferation by controlling the export of materials, equipment and technology that can be used to manufacture nuclear weapons. NSG member states, including Russia, comply with the NSG guidelines which have been implemented into the national legislation and are binding at the corporate level. In particular, these guidelines are reflected in the projects of Rusatom Service Company.

Rusatom Service nowadays implements an immense project on assessment of the nuclear infrastructure for construction of Center for Nuclear Technology Research and Development. Conclusion of the Contract between Rosatom and ABEN. The scope of work:

- Analysis of current state based on existing national regulatory documents and expert interviews
- Elaboration of technical report, containing information and recommendations for 19 elements of nuclear infrastructure
- Preparation of the Roadmap for measures to develop national nuclear infrastructure in Bolivia including development of Export Control system.

B. Contract Screening

A great work has been done regarding the contract screening to verify export control issues. This work was divided into 3 stages corresponding to the stages of the project implementation. All 'bottlenecks' were identified and determined as those subject to export control measures. All necessary export control procedures were initiated.

V. THE EMERGENCE OF SINGLE DIGITAL PLATFORMS IN THE FIELD OF EXPORT CONTROL

A. Reasons for Developing and Problem-Solving of Single Digital Platforms

The emergence of single digital platforms in the field of export control has now been noted at the state level in a number of states [1]. Within the framework of these platforms, the automation of procedures has significantly reduced the time necessary for providing services, its implementation also had a considerable economic effect and minimized the risks of export control violations by the participants of foreign trade activities.

In 2018, Rusatom Service launched a project in order to create a single digital export control platform named DIRECT.Compliance. The name was chosen not by chance – "DIRECT" in this case is an acronym ,which means Digital Intelligent Rosatom Export Control Tool.

The goal of the project is to create a unified infrastructure within the framework of export controls in order to minimize the risks of foreign economic activity and transactions.

By creating a common information space within the enterprise-integrator of the nuclear industry we can structure and systematize the export control work, as well as to ensure the compliance by employees of the enterprise with the provisions of an industry order for the organization of export control and the internal program of export compliance.

Systematization of the work of sectoral enterprises allows to simplify the interaction between the enterprises of the Rosatom State Corporation on the expertise of draft external economic contracts, additional agreements and additions to them. This is especially true today, when the export control function might be outsourced, and therefore, the customer must control the execution of the tasks assigned to the service provider. In the industry, it is Rusatom Service that created the product line dedicated to export control, and regularly seeks improvements in this area, expanding the client network and working on daily qualitative and quantitative additions to the services provided.

Many participants of foreign economic activity in Rosatom State Corporation often face certain administrative barriers while coordinating and implementing foreign economic transactions. The reason for such problems may be the lack of a single consolidated export control platform, a complicated scheme of interaction between participants of foreign economic activity and federal executive bodies, the human factor (the presence of errors in documents, low qualification of personnel responsible for export control), the lack of a well-designed internal export control program. [2]

Technically, DIRECT Compliance is a digital platform where managers of Rusatom Service JSC can develop documents that relate to export control issues, send them to authorized bodies as well as directly consult with officials concerning different nuances. Moreover, our clients have access to this system, and they have an opportunity to follow all stages of gaining authorizations for export activities online. [3]

B. Short Movie About the 'DIRECT.Compliance'Russian platform (duration 2.5 minutes)

Development and implementation of a single digital export control platform is essential for the development of international cooperation, as well as the optimization of export control activities. The created platform will allow not only to improve sectoral export control, but also to implement inter-sectoral projects, as export control issues affect chemical and biological products, missile production, the military-industrial complex, as well as other advanced industries. At present, there is no system in the Russian Federation providing all the functionality that was incorporated into this platform. The issue of necessary digital interaction on a national scale is stated by the Government of the Russian Federation and other participants of foreign economic activity. [4]

VI. CONCLUSION

Digital export control service platform 'DIRECT. Compliance' is a multiplatform tool for effective interaction of subjects of foreign economic activity with regulatory authorities.

This multiplatform tool is aimed to create a unified infrastructure for export control in order to minimize the risks of export control violations during foreign economic activities. The implementation of the 'DIRECT.Compliance' platform in enterprises will result in the removal of administrative barriers in the field of export control from the regulatory authorities. A potential direction for the development of a single digital platform can be integration with similar existing platforms in Rosatom partner countries, which will significantly simplify the interaction between countries in the field of export control, as well as contribute to strengthening the nuclear non-proliferation regime. A single digital platform within a single enterprise creates the basis for further scaling beyond the nuclear industry and creating a sectoral and national software package that interacts between traders, relevant ministries and state corporations, federal executive bodies and other parties involved.

In conclusion, we would like to mention that the Russian Federation has harmonized national legislation with international standards. Our country provides full compliance with guidelines of the international EC-regimes such as NSG regime.

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Application of Probabilistic Safety Assessment to Justification of Design Solutions related to Ultimate Heat Sink Protection from External Hazards

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I. Introduction

To implement modern safety requirements, the entire life cycle of a nuclear power plant is accompanied by a number of serious studies confirming the safety of construction and operation. A building and operating permit can only be obtained if safety has been proven and an examination of state regulatory authorities has been completed. Probabilistic safety assessment (PSA) occupies an important place among these studies.

The reliability issue of nuclear power plants was raised from the very beginning of their development. Nevertheless, in spite of the fact that the first nuclear power plant was launched in Obninsk in 1954, the first large-scale work devoted to PSA was published in the USA only in 1975 [1]. In Russia, probabilistic methods for calculating the reliability of nuclear power plants have been used since the late 60s of the 20th century. Starting with OPB-88/97 [2], the presence of a PSA was required as part of the design documentation submitted to obtain a license from the Gosatomnadzor of Russia [3].

Fukushima Daiichi nuclear disaster occurred due to external hazard. Severity of the tsunami consequences can be explained by loss of ultimate heat sink (LUHS). The ultimate heat sink of a nuclear power plant is a complex cooling system to which residual heat removed during a variety of both normal and emergency operating modes.

Fukushima Daiichi accident brought innovations to the process of nuclear power plants design including PSA. Consideration of external hazards during the design (e.g. in PSA) entails special requirements for systems, structures and components (SSC) of a nuclear power plant [4, 5, 6].

The objective of the current study was the adequacy of protective measures implemented in the design to withstand the influence of external hazards that entail the LUHS. The aim of the study is to confirm the design decisions applied at a modern nuclear power plant in relation to external hazards associated with the LUHS, and to estimate the probabilistic safety values associated with this.

To achieve this goal, the following tasks were solved:

- SSC of a nuclear power plant with a VVER-1200 reactor installation related to the ultimate heat sink were described;
- State-of-the-art methods used to evaluate probabilistic safety values were examined;
- Requirements for SSC associated with the influence of external hazards were analyzed;
- Nuclear fuel damage frequencies due to external hazards entailing violation of ultimate heat sink were assessed.

II. Theory and Methodology

PSA is a means of identifying potentially vulnerable aspects in the design and control of nuclear power plant units. It is a complex, comprehensive safety analysis, during which probabilistic models are developed to determine end states with damage to sources of radioactivity. End states of nuclear power plant with exceeding the established limits on emissions of radioactive products, radiation impact on the population and environment are determined estimating values of probabilistic safety indicators [7].

According to international and the Russian regulating and methodical documents [8, 9, 10] the PSA is subdivided into several levels. Each of them is intended for own purposes and possible use of results.

Reactor and spent fuel pool were taken as sources of radioactivity in current study.

In the PSA of level 1 realistic probabilistic models are developed for definition of end states with damage of sources of radioactivity. Values of frequencies of damage of nuclear fuel are estimated, recommendation about increasing the safety level of the NPP are developed.

For achievement of goals of PSA-1 it is necessary to complete the following tasks:

- Collecting and preparation of an original information that is necessary for characteristic of the power unit;

- Selection and grouping of plant operational states of the power unit;
- Selection and grouping of initiating events (IE);
- Modeling of the accident sequences;
- System analysis;
- Data analysis on frequencies of the initiating events and reliability of the equipment;
- Human reliability analysis;
- Development of a logical probabilistic model of the power unit;
- Calculation of nuclear fuel damage frequency of the power unit of nuclear power plant;
- Analysis of uncertainty, sensitivity and importance;
- Assessment of the security level of the power unit and analysis of results of PSA.

Depending on the range of recorded initiating events, PSA is classified into internal IE or full-scale. Internal events include all possible failures of NPP equipment related to probabilistic nature of failure and also human errors. Full-scale PSA level 1 also takes into account:

- On-site fires;
- On-site flooding;
- Other on-site threats;
- Seismic threats;
- External hazards except seismic threats.

According to the Russian regulatory and technical documents [8], the target safety parameter of the nuclear power plant in relation to the probability of a severe accident at the interval of one year is 10^{-5}.

Hazards usually associated with loss of ultimate heat sink were taken as initiating events in current study:

- Algae;
- Frazil ice;
- Pack ice;
- Oil spill.

Defense-in-depth principle is based on the use of a system of physical barriers to hinder the spread of ionizing radiation and radioactive substances into the environment. Those barriers and system of technical and organizational measures to protect them and preserve their effectiveness ensure the safety of the NPP.

The requirements for systems and components in a nuclear power plant depend on the events they must withstand.

The ultimate heat sink and associated heat transfer systems are designed to remove heat from the reactor in all design operating modes of the plant.

The Baltic Sea water is used as a source of technical water supply and a final heat swallower at considered nuclear power plant. Water lines of cooling water (channels, tunnels) connect the final heat absorber with consumers of sea cooling water of the power unit. All cooling water systems are made according to direct-flow scheme with single circulation of seawater through heat exchange equipment.

The following cooling water systems are applied in the design of considered NPP:

- Main cooling water system intended for cooling water supply and heat removal to the final absorber from turbine in all normal operation modes;
- Auxiliary cooling water system designed for heat removal to final absorber from equipment not important for safety;
- Cooling water system of essential consumers is intended for heat removal to the ultimate heat sink from heat exchangers of the intermediate circuit system of essential consumers. The system is operated in all modes of unit operation including extended design modes.

RiskSpectrumPSA was used as a tool for estimating of probabilistic safety values including development of main steps of PSA: fault tree analysis, event tree analysis and their calculation.

According to a methodology of the PSA the main steps for assessment of probabilistic safety values were executed. Plant operational states were taken according to reactor plant operating instructions, scheduled maintenance regulations and results obtained by performing this task in PSAs for similar nuclear facilities.

The accident sequences were analyzed based on current design. Event trees were elaborated for purpose of analysis. Figure 1 presents results of development of event tree for initiating event – “Algae”. Due to excessive dimensions of this tree it was reduced using transfer event trees (starting with “>” symbol in Consequences).

Several functional events are used to model the safety functions. “00_POS” is used for modeling of dividing into different plant operational states. “FORWARD” shows all protective measures implemented in the design to ensure the direct way of water intake (in case of algae it is rotating screens with submersible pumps, grippers and trash racks). Functional event “REVERSE” models all the necessary actions completed for switching of water intake in reverse way (personnel actions and equipment operation).

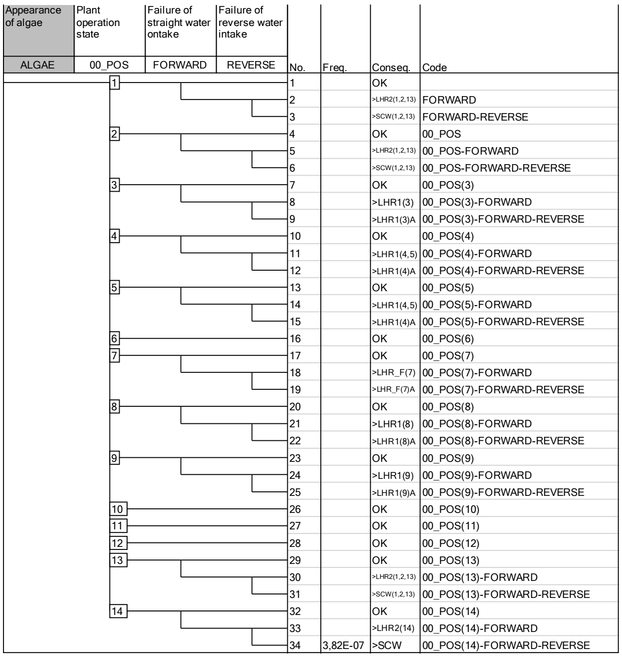

Figure 1. Event tree for initiating event "Algae"

Systems analysis was accomplished and fault trees in the software of RiskSpectrumPSA for the selected safety features were constructed. Fault tree were used as inputs for function events from event trees. Following steps were completed during system analysis:

- Definition of system purpose and PSA-related safety functions;
- System boundaries definitions;
- Identification of initial condition of the system for analysis;
- Appointment of modeling assumptions;
- Failure modes and effects analysis;
- Analysis of dependencies and connections with other systems;
- Human reliability analysis;
- Common cause failures analysis;
- Fault tree development.

III. RESULTS

Table I presents results of final quantification. Following abbreviations were used to mark possible types of nuclear fuel damages:

- CD – core damage during power operation;
- CDS – core damage for the events happening during outage modes;
- FDS – damage of fuel in spent fuel pool for the events happening during outage modes.

Table II contains results of quantification of frequency of loss of ultimate heat sink due to different initiating events.

TABLE I. RESULTS OF FINAL QUANTIFICATION

IE	IE frequency, 1/year	Fuel damage	Value, 1/year	Conditional probability	Contribution to total nuclear fuel damage frequency, %
Algae	$1{,}05\cdot10^{-1}$	CD	$1{,}87\cdot10^{-12}$	$1{,}78\cdot10^{-11}$	0,0001
		CDS	$2{,}88\cdot10^{-10}$	$2{,}74\cdot10^{-9}$	0,0229
		FDS	$1{,}21\cdot10^{-12}$	$1{,}15\cdot10^{-11}$	0,0001
		Sum	$2{,}91\cdot10^{-10}$	$2{,}77\cdot10^{-9}$	0,0231
Frazil ice	$4{,}17\cdot10^{-2}$	CD	$1{,}57\cdot10^{-11}$	$3{,}76\cdot10^{-10}$	0,0012
		CDS	$1{,}29\cdot10^{-9}$	$3{,}09\cdot10^{-8}$	0,1024
		FDS	$1{,}56\cdot10^{-10}$	$3{,}74\cdot10^{-9}$	0,0124
		Sum	$1{,}46\cdot10^{-9}$	$3{,}51\cdot10^{-8}$	0,1160
Oil spill	$3{,}70\cdot10^{-5}$	CD	$3{,}62\cdot10^{-11}$	$9{,}78\cdot10^{-7}$	0,0029
		CDS	$2{,}11\cdot10^{-9}$	$5{,}70\cdot10^{-5}$	0,1675
		FDS	$1{,}03\cdot10^{-12}$	$2{,}78\cdot10^{-8}$	0,0001
		Sum	$2{,}15\cdot10^{-9}$	$5{,}80\cdot10^{-5}$	0,1704
Pack ice	$8{,}77\cdot10^{-3}$	CD	$4{,}75\cdot10^{-18}$	$5{,}42\cdot10^{-16}$	0,0000
		CDS	$5{,}33\cdot10^{-19}$	$6{,}08\cdot10^{-17}$	0,0000
		FDS	$1{,}79\cdot10^{-15}$	$2{,}04\cdot10^{-13}$	0,0000
		Sum	$1{,}80\cdot10^{-15}$	$2{,}05\cdot10^{-13}$	0,0000
Summary	–	–	$3{,}90\cdot10^{-9}$	–	0,3095

TABLE II. FREQUENCY OF LUHS DUE TO DIFFERENT IE

IE	Probability of equipment failure	Frequency of LUHS, 1/year
Algae	$8{,}64{\cdot}10^{-7}$	$9{,}07{\cdot}10^{-8}$
Frazil ice	$5{,}52{\cdot}10^{-6}$	$2{,}30{\cdot}10^{-7}$
Pack ice	$1{,}00{\cdot}10^{-7}$	$8{,}77{\cdot}10^{-10}$
Oil spill	$1{,}18{\cdot}10^{-2}$	$4{,}43{\cdot}10^{-7}$

IV. DISCUSSION

All safety function of nuclear power plant can be presented using the scheme from Figure 2 where marks mean following [11]:

- F1A – safety functions which are needed for transfer of nuclear power plant to controlled state in case of design basis condition events;
- F1B – safety functions, which are needed for transfer of nuclear power, plant to safe state in case of design basis events. If safe state is reached less than in 24 hours, then F1B safety function shall maintain safe state at least for 24 hours after initiating event;
- F2 – safety functions, which are needed for maintenance of safe state in time interval 24 – 72 hours in case of, design basis event. Also F2 safety functions are involved in nuclear power plant transfer to safe state in case of design extension events.

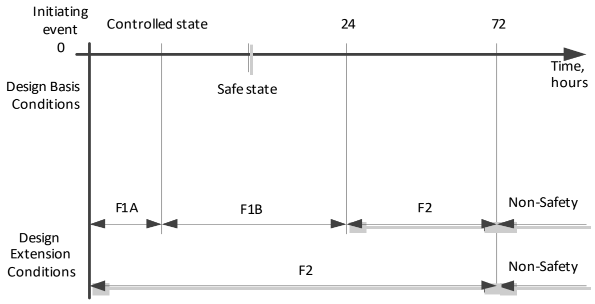

Figure 2. Safety functions distribution

Approach for implementation of requirements can be less strict for F2 safety functions than for F1A and F1B. For example, single failure criteria, additional power supply, physical separation of safety trains and automatic actuation can be omitted if proved by safety analyses according to international design experience.

In current study design extension conditions connected with external events (DEC-C) are considered. For such events, it is necessary to estimate their frequency of occurrence. Events were assumed belonging to DEC-C if their frequency of occurrence is estimated less than 10^{-5} 1/year. In this study not the frequency of initiating events are taken but combination of events leading to LUHS (initiating events and equipment failure).

According to results of loss of ultimate heat sink frequencies quantification in Table 3 all events can be defined as DEC-C conditions. It means that requirements implemented to such system during design process can be taken as to F2 safety function category.

Conditional probability describes the frequency of fuel damage due to some initiating events. Conditional probabilities presented in Table I justify sufficiency of implemented design solutions for protection against external hazards leading to loss of ultimate heat sink.

Such value as contribution to the total nuclear fuel damage frequency represents which part is contributed by the considered initiating event in the total nuclear fuel damage frequency. Contributions presented in Table I (summary less than 0,3 %) allow to treat considered initiating events as low contributing.

V. CONCLUSION

The results show that the applied protective measures against external hazards leading to the loss of ultimate heat sink are sufficient for safety in accordance with international and Russian regulations. It shall be mentioned that used hazard frequencies are estimated in conservative manner. Such estimation could lead to conservatism of presented in current study scenarios frequencies estimation. Therefore uncertainties related with these LUHS frequencies values are covered to some extent.

Chosen design solutions allows to reduce time of nuclear power plant construction and in the same time meet all modern safety requirements.

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Visualization of radioactive substances using freely moving gamma-ray imager based on Structure from Motion

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I. Introduction

Technology used to measure and identify the location and distribution of radioactive substances is important for decommissioning work sites at nuclear power plants. The Fukushima Daiichi Nuclear Power Station (FDNPS) operated by Tokyo Electric Power Company Holdings, Inc. went into meltdown after the large tsunami caused by the Great East Japan Earthquake on March 11, 2011. Thus, it is necessary to improve the measurement method of the distribution of radioactive substances scattered in such work environments to reduce the dose rate for workers and to develop decontamination plans.

The main radionuclide that contributes to the increase in the dose rate at the work site of FDNPS is radioactive cesium, and the location of the radioactive substances has been roughly identified by measuring the gamma rays emitted using handheld survey meters. However, since conventional survey meters cannot determine whether gamma rays have come from a particular direction, it is difficult to specify the detailed location and distribution of radioactive substances that have emitted gamma rays. Particularly at decommissioning work sites, because radioactive substances are present on many different structures, such as equipment and rubble, it is necessary to determine the distribution of the radioactive substances three-dimensionally (3D). Furthermore, it requires a long time to measure a wide area using a survey meter, and there is a risk that workers may be exposed during this time period.

Here, several techniques for determining the distribution of radioactive substances using gamma-ray imagers, such as a pinhole-type camera, coded aperture-type camera, and Compton camera, have been introduced. For the pinhole-type camera and Compton camera, several demonstrations have already been conducted in the damaged building of the FDNPS [1-3]. Furthermore, in recent years, a method of acquiring images of the radioactive substances in 3D using freely moving devices has been demonstrated by combining these gamma-ray imagers and Simultaneous Localization and Mapping (SLAM) technology. For instance, a system that combines a Microsoft Kinect sensor and a gamma-ray imager has been demonstrated [4]. A 3D structural model of the experimental environment is constructed by using Kinect while freely moving, and self-positioning estimations are performed by the device. At the same time, the position of the radioactive substance is visualized in 3D by projecting the image of the radioactive substance acquired by the gamma-ray imager onto the 3D structural model. The Microsoft Kinect sensor also provides RGB colors on the 3D point cloud data (PCD) model of the environment. Because it is low-cost device compared to light detection and ranging (LiDAR) devices and can use RGB information, it is advantageous for identification of an object that cannot be identified by LiDAR, such as the type of material. However, it has a disadvantage: the Kinect has a measurable distance that is short by several meters. It takes a considerable time to reconstruct a large environment that contains large objects, such as the damaged buildings of the FDNPS.

In this report, we propose 3D radiation imaging technology that uses freely moving devices by combining Structure from Motion (SfM) and a Compton camera. SfM is a technique to simultaneously perform a self-position estimation and construct a 3D structural model of an environment using many photographs acquired by a moving optical camera [5]. Ordinary digital cameras can be used and are low cost compared with the Kinect. Furthermore, a wider range of 3D structural models can be reconstructed because performing the photography. The methodologies for visualizing radioactive substances in 3D and the results obtained by combining the SfM and the Compton camera are provided.

II. 3D Gamma-ray Imaging Using a Compton Camera While Freely Moving Based on SfM

A. Methodology

A method of identifying in 3D the position of a radiation source using a Compton camera has already been demonstrated via simulations and experiments [6,7]. Compton cones are projected onto a 3D space, and a radiation source is found at the part where Compton cones overlap in the 3D space. We have also developed a method for projecting images of radioactive substances measured with the Compton camera onto a 3D structural model of a work environment reconstructed by using multiple photographs of the work environment [8]. However, in this method, when measuring a radioactive substance using the

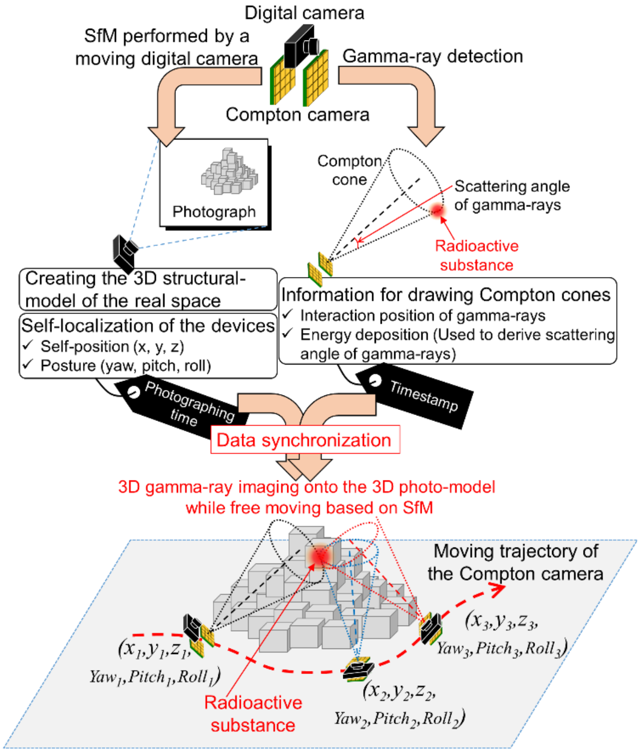

Fig. 1. Schematic of 3D visualization of radioactive substances using a freely moving Compton camera based on SfM. The positions where the Compton camera detects the gamma ray are estimated by SfM using the digital camera; the timestamp of the gamma-ray events and the photographing time of the photographs are synchronized. Compton cones from each position are projected on the 3D model of the experimental environment.

Compton camera, the Compton camera is fixed, and many measurements are performed. When measuring a target object from many directions using the Compton camera, it is necessary to measure each located point and the posture of the Compton camera using some method. If these pieces of information are not obtained, the projection direction of the Compton cone cannot be determined.

Then, we also applied the self-positioning and posture determination of the Compton camera using SfM for the measurement of a radiation source from multiple viewpoints [9]. However, when performing gamma-ray measurements, the Compton camera is fixed at each located point. It is more efficient to acquire the data for visualization of radioactive substances via gamma-ray measurement while freely moving the Compton camera and obtaining photographs to build a 3D structural model of the work environment simultaneously.

Here, based on SfM, we introduce a method for 3D identification of the position of radioactive substances using a freely moving Compton camera, as shown in Fig. 1. This method was performed by combining a digital camera (optical camera) and a Compton camera. By carrying out the gamma-ray measurement while freely moving in the experimental environment, data for creating an image of the radiation source on a 3D structural model can be obtained.

This method includes two processes: SfM using a digital camera and gamma-ray measurements using a Compton camera. The 3D structural model of the experimental environment was reconstructed using photographs taken from moving the digital camera, and information of the camera position and posture when each photograph was taken was estimated by SfM. Furthermore, because the information of the photographing time is provided for each photograph, the position and posture of the digital camera combined with the Compton camera at that time can be determined.

On the other hand, a timestamp was also added to gamma-ray events measured by the Compton camera. Specifically, time information is provided to the interaction position of the gamma rays at the sensor of the Compton camera and the energy deposition to the sensor. This information is used to calculate the scattering angle of the gamma rays necessary for drawing the Compton cone. Here by synchronizing the time when the photograph was taken and the time when the gamma-ray event is measured by the Compton camera, the direction in which the Compton cone is projected can be determined at each measurement time, and the radiation source can be found where the Compton cones overlap on the points of the 3D structural model constructed by the SfM.

Our method takes into account that the field of view (FOV) of the Compton camera overlaps by measuring the target region from multiple viewpoints. In the measurement environment, the area where the FOV overlaps due to measurements from multiple viewpoints results in a longer measurement time. Therefore, even if the radioactivities of the radioactive substances are the same, the image of the radioactive substance is drawn to be stronger than the area where the overlapping of the FOV is small. Therefore, for each area in the 3D structural model, the total time measured from multiple viewpoints was calculated, and the intensity of the image of the radioactive substances was corrected using this value. Specifically, the value set by the Compton cone projection at each point in the PCD of the experimental environment was divided by the integral time at which each point was measured.

B. Demonstration

A demonstration experiment for visualizing the radiation source was carried out using the combined equipment of a Compton camera and a digital camera. Figure 2 shows a photograph of the equipment used for the demonstration test, which is the combination of a Compton camera and a digital camera (DSC-RX100M3, SONY, Japan). The Compton camera consists of two gamma-ray sensors, i.e., a scatterer and absorber, as shown in Fig. 3. These sensors employ a Ce-doped GAGG ($Gd_3Al_2Ga_3O_{12}$) scintillator coupled with a multi-pixel photon counter (MPPC; Hamamatsu Photonics K.K.) [10,11]. The scattering angle of gamma rays on the scatterer were estimated by using the numerical formula for Compton scattering. In addition, using this scattering angle, a Compton cone was virtually drawn in front of the detector, and a radioactive

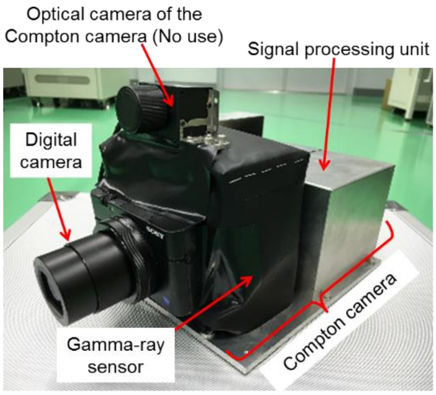

Fig. 2. Photograph of the equipment used for 3D radiation imaging. A digital camera was installed in front of the Compton camera.

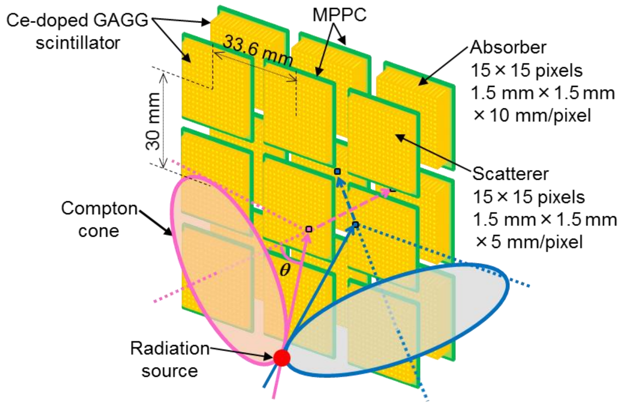

Fig. 3. Schematic of the gamma-ray sensor of the Compton camera and the image reconstruction of radioactive substances using the back-projection method

substance could be found at the intersection of many Compton cones. The size and weight of the Compton camera are 182 mm × 199 mm × 156 mm and 2.5 kg. This Compton camera is a customized version of the GAMMA Catcher supplied by Chiyoda Technol Corporation, Japan, and was fabricated based on the technology from the handheld Compton camera jointly developed by Waseda University and Hamamatsu Photonics K.K. [12,13].

Figure 4(a) shows a 3D structural model of the PCD reconstructed using the photographs taken using the moving digital camera installed on the front of the Compton camera. The prepared photographs were taken at 1-s intervals. The total number of photographs used for SfM was 287. The ceiling was removed to make it easier for readers to understand the experimental environment. These reconstructions were performed by PhotoScan software [14]. The color dots displayed in the 3D structural model indicate the photographing point of the digital camera estimated by SfM via PhotoScan software, and the operator walked while taking photographs along this trajectory. At the same time, the Compton camera measured the gamma rays. Here, to perform a visualization test of a radiation source, a ^{137}Cs radiation source of 10 MBq was placed in a shoe box.

Next, drawing an image of a radiation source is described. As shown in Fig. 1, the detection time of each gamma ray is synchronized with the photographing time of the digital camera, and the self-position and posture information of the Compton camera when the gamma rays were detected were estimated by SfM. Thus, the projection direction of the Compton cone can be automatically determined. Figure 4(b) shows the imaging result for the radiation source. Figure 4(c) also shows the enlarged view of the 3D photo model of the area around the shoe box. In addition, a map showing how long each region was measured is shown in Fig. 4(d); the total time that each point in the PCD was in the FOV was calculated. In the radiation images shown in Figs. 4(b) and (c), the intensity of each point composing the 3D PCD-model was corrected using the measurement time.

In the image reconstruction, gamma-ray event data used for image reconstruction were selected to visualize the ^{137}Cs-radiation source, which emits 662-keV gamma rays. The selected energies were set to 625 keV ≤ $E_s + E_a$ ≤ 725 keV and 10 keV ≤ E_s ≤165 keV, where E_s and E_a are the energy depositions in the scatterer and absorber, respectively. Compton cones were drawn on the 3D structural model using these gamma-ray event data. The total number of Compton cones for using the drawn the radiation image was 232. In addition, the FOV of the Compton camera was ±90 degrees.

The image intensity near the shoe box containing the ^{137}Cs source was higher than that for the surrounding area, and the presence of the radiation source can be estimated to be in the vicinity of the shoe box. The accuracy of determining the location of the radiation source, i.e., the distribution of the radiation image, was approximately 60, 50, and 45 cm in the *X*, *Y*, and *Z* directions when the intensity threshold was set to 0.7. By using this figure, it can be visually understood that a radiation source is present in inside the shoe box. The source image is displayed with a high intensity on the right side of the shoe box, which is consistent with the source being placed along the right wall inside the shoe box.

Unlike the conventional fusion of 3D structural modeling technology for experimental environments that involves photographs and radiation imaging technology, this technology realized visualization of a radiation source by acquiring data while freely moving the devices.

III. Conclusions

We developed a three-dimensional (3D) image reconstruction method for radioactive substances by integrating a Compton camera with SfM. By freely moving the experimental environment while taking photographs with a digital camera, a 3D structural model of the experimental environment could be reconstructed using photographs. It is also possible to estimate information about the camera's own position and posture when taking each photograph. Because the Compton camera and the digital camera are combined, information from the Compton camera is also automatically obtained. By synchronizing the time when the gamma rays were detected on the Compton camera with the photographing time of the digital camera, the Compton camera's self-position and posture information when the gamma rays were detected could also be estimated. This makes it possible to determine the projection direction of the Compton cone at each position on the movement trajectory. In this demonstration, we succeeded in visualizing a ^{137}Cs-radiation source by measuring an experimental environment while freely moving using these cameras. This method will allow successful construction of a map to visually recognize the positions of radiation sources inside the FDNPS.

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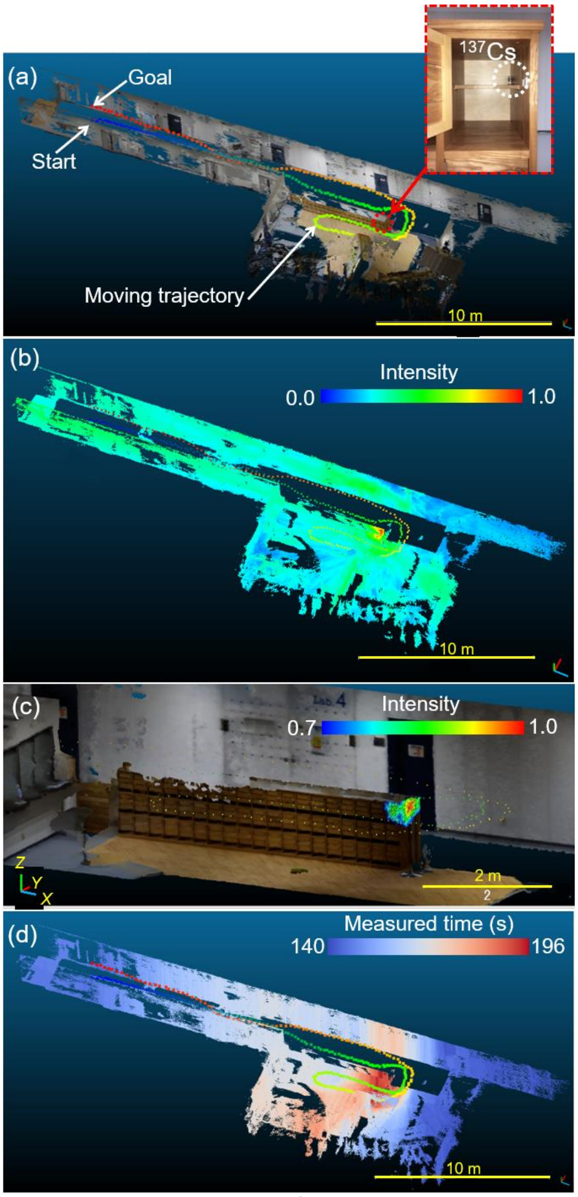

Fig. 4. (a) 3D structural model of the experimental environment created by SfM. The moving trajectory of the imaging device is also shown. The color of the points shows the elapsed time from the start of the measurement. A ^{137}Cs radiation source was installed inside the shoe box. A series of photographs were taken by the moving digital camera to reconstruct a 3D model of the experimental environment. (b) Radiation image drawn on the 3D model. (c) Enlarged view of the area around the shoe box. The high-intensity image appeared on the shoe box that contained the ^{137}Cs radiation source. Photographs of the experimental environment were pasted on the 3D model for texture. (d) Integral time included in the FOV of the Compton camera in each region.

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Correlation between Heat-Mass Transfer, Chemical Reactions and Phase Transformations in Corium Melt Localization Devices during Severe Nuclear Power Plant Accidents

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I. INTRODUCTION

Nuclear power plant (NPP) safety has a number of unsolved issues these days for example: the events surrounding ‘Three Mile Island’, ‘Chernobyl’ and ‘Fukushima’ NPPs [1]. So, along with attempts to avoid such accident efforts are being made to mitigate their consequences. This is supposed to be done through localization of high-temperature corium which is a mixture of nuclear reactor components such as uranium, plutonium and zirconium oxides (Corium-A) and construction materials like steel and concrete (Corium-R). In order to achieve this, both internal and external cooling traps were developed (molten Corium-A+R).

II. INTERNAL AND EXTERNAL CORE TRAPS

Internal core traps are based on cooling the bottom of the reactor vessel with water, which requires free water circulation through the reactor shaft. However, during severe NPP accidents the bottom of the reactor vessel undergoes significant thermal and thermomechanical loads, so this concept is only applicable for medium and low power reactors. It was first embodied at ‘Loviisa’ NPP in 1989 (Fig. 1).

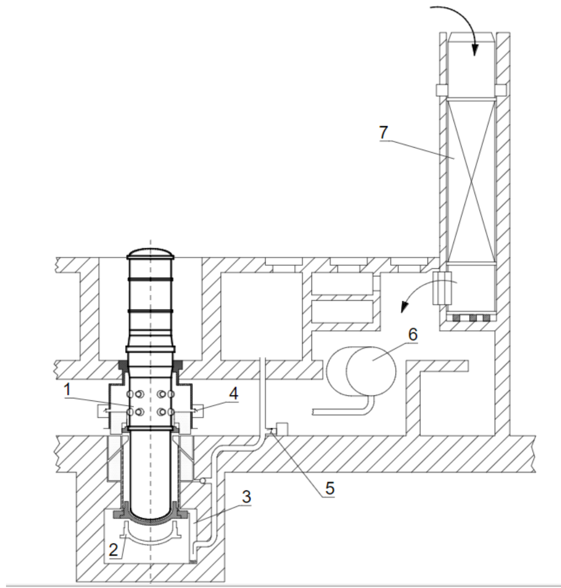


Figure 1. Corium retention system at Loviisa NPP with VVER-440/213: *1* - reactor vessel, *2* - reactor vessel screen, *3* - filter, *4* - exhaust valves, *5* - intake valves, *6* - steam generators, *7* - ice condenser.

Most of the new-developed reactors are high power ones, so they can only rely on external core traps [2]. The development of external core traps is profitable in many aspects: first, its architecture is based on multi-level protection; it is going to serve as an additional barrier on the melt propagation path and to reduce the loads on the containment in case of reactor vessel breakdown; second, it allows more space for mechanical designers, therefore various of core traps are possible.

These days, core traps are realized at European EPR (Fig. 2) and Russian AES-2006 (Fig. 3) reactors [3].

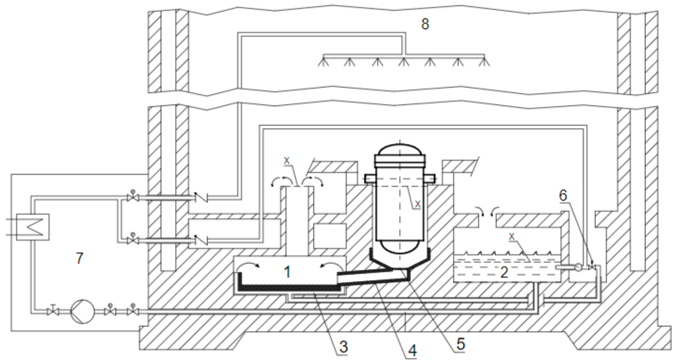


Figure 2. The scheme of the device for localizing the melt of the EPR reactor:
1 - spreading box, *2* - reloading pool, *3* - water supply to core trap, *4* - inclined channel with a protective layer, *5* - smelting plug, *6* - passive bay device, *7* - heat removal systems from containment (2 channels), *8* - sprinkler devices, *x* - water level in case of active injection of water into the spreading box.

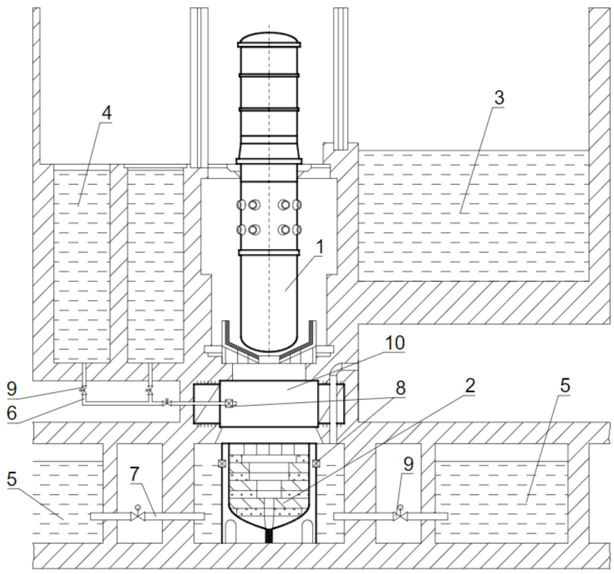


Figure 3. Crucible-type melt localization device for AES-2006:
1 – reactor, *2* – core trap crucible, *3* – fuel pool, *4* – internal inspection chamber, *5* – sump tanks, *6* – water supply pipes to the console farm and to the surface of the corium, *7* - pipelines for supplying water to the heat exchanger ULR, *8* - valves for passive water supply with a thermomechanical element, *9* - valves, *10* - truss console.

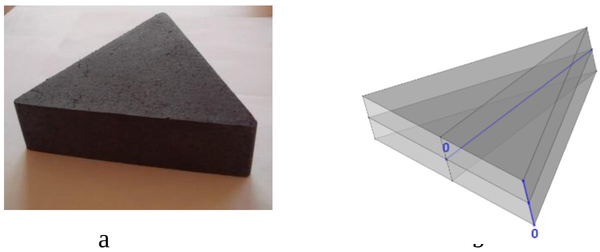
Despite of all the differences between these schemes, their disadvantages are common. The interaction of high-temperature corium and core trap sacrificial material (temperature-consuming and radionuclide immobilizing material) has not yet been studied well. This interaction should lead to a corium temperature decrease, an increase of heat-radiating surface, a reduction of oxide melt component density and oxidation of zirconium which leads to a decrease of hydrogen production in steam-zirconium reaction [4].

An analysis of the data showed that during the interaction between corium and sacrificial material in the first stages of an accident, chemical reactions have little effect, then with the formation of corium, given its high temperature and chemical aggressiveness, they begin to significantly affect heat and mass transfer processes and phase transition processes. At the same time, feedbacks appear that determine the conditions and rates of the chemical reactions themselves [5].

III. Model and Results

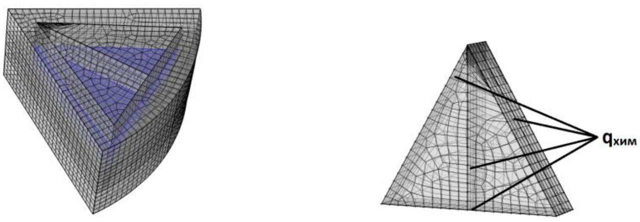
After 20 minutes, the thin crust of the corium initially formed on the surface of the sacrificial material begins to melt, but due to its low thermal conductivity, this process lasts 100 minutes. About the same amount of time hematite is melted.

Model description is presented in the Fig. 4, Fig. 5, Fig. 6 and table 1.



a b

Figure 4. The structural element of the sacrificial material based on Fe_2O_3 and Al_2O_3 (a) and its section design (b).



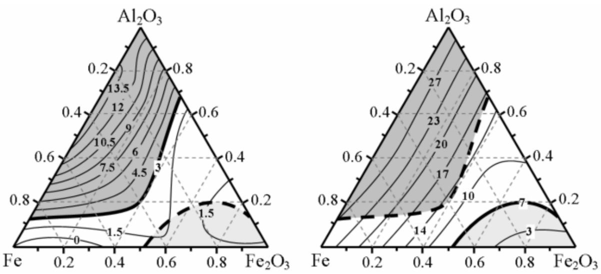
(a) (b)

Figure 5. The mesh of the final elements of the sacrificial material, taking into account the surrounding corium melt (a), heat sources at the boundaries (b).

TABLE I. Heat Sources in the Volume of the Sacrificial Material

Parameter	**Value**			
τ, s	0	3600	7200	1080
Q, MW	1,84	1,465	1,29	1,18

Note: the calculated heat flux at the border of corium and hematite is taken q = 14900 W; initial hematite temperature T_{0h} = 300 K; initial temperature of the corium T_{0c} = 3000 K.

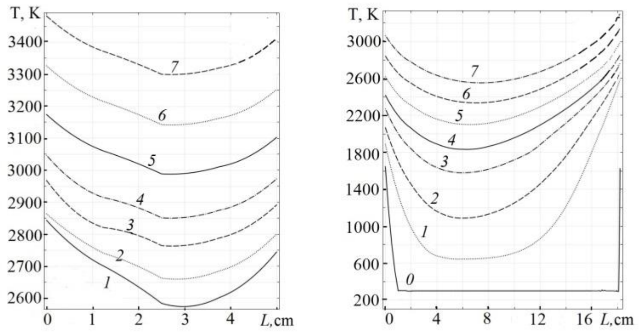


a b

Figure 6. The mesh of the final elements of the sacrificial material, taking into account the surrounding corium melt (a), heat sources at the boundaries (b)

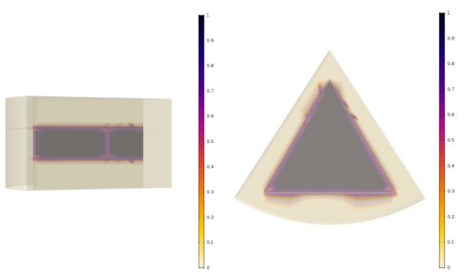
Fig. 6 show data of thermodynamic calculations of the interaction of the core melt of the nuclear reactor and the sacrificial material (a - the amount of produced gases; b- integral heat losses during the interaction of the core melt with the sacrificial composition).

Using this model, the following results were obtained Fig. 7 and Fig. 8.

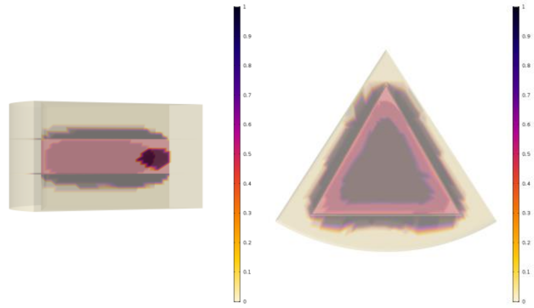


a b

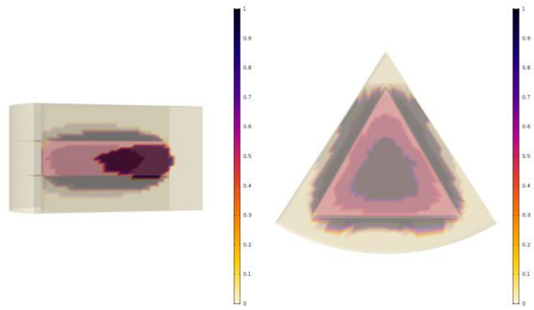
Figure 7. Change in temperature distribution over the thickness (a) and height (b) of an element of a sacrificial material during its interaction with the corium melt over time: *0*- 0 min, *1*- 5 min, *2*- 10 min, *3*- 20 min, *4*- 30 min, *5*- 50 min, *6*- 80 min, *7*- 120 min.



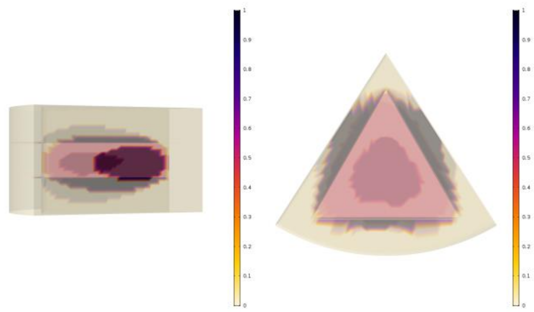
a) τ =0 min



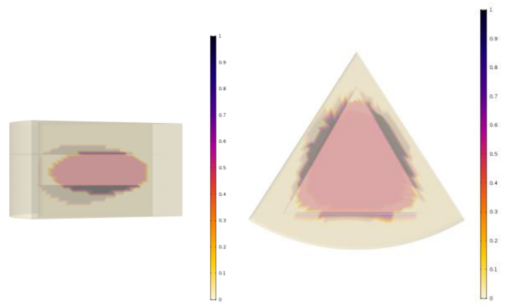
b) τ =5 min



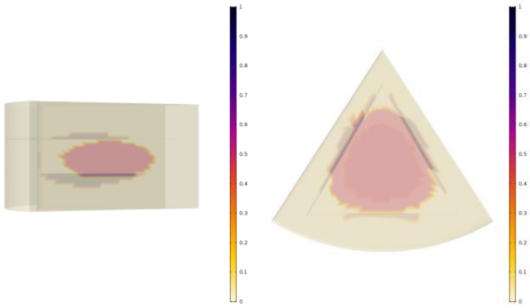
c) τ =10 min



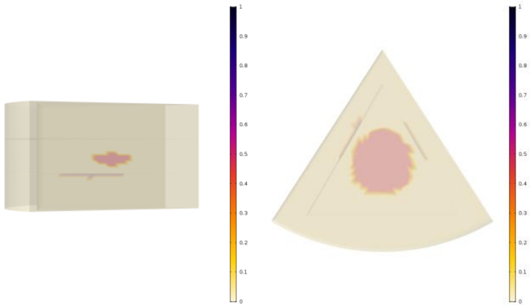
d) τ =20 min



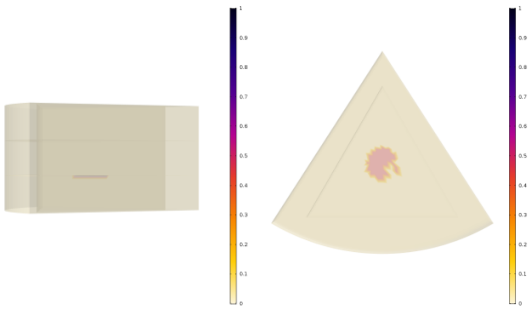
e) τ =40 min



f) τ =60 min



g) τ =100 min



h) τ =120 min

Figure 8. Change in the ratio of the areas of solid and liquid phases in the structural element of the sacrificial material (in thickness and on the surface) and the surrounding corium during their interaction in time (a-h).

Fig. 7 and Fig. 8 show the thermal and phase-kinetic characteristics of the dynamics of the processes of interaction of the high-temperature corium melt with the structural element of the sacrificial material. Analysis of the graphs of the temperature changes over time over the calculated cross sections of the sacrificial material (Fig. 7) and changes in the ratios of the regions of the solid (dark area) and liquid (light area) phases in the structural element of the sacrificial material and the surrounding corium during their interaction in time (Fig. 8) showed the following.

In the initial stage of interaction, a thin layer of crystallization crust from the corium melt is formed on the surface of the sacrificial material. Its low thermal conductivity reduces the thermal effect of chemical reactions. With further heating due to heat exchange with the surrounding high-temperature corium melt, this crust begins to melt, as does the sacrificial material itself. At the same time, a significant increase in the influence of chemical reactions on the processes of heat-mass transfer and phase transitions in it is observed, the feedback with which in turn determines the conditions for the course of the chemical reactions themselves.

Conclusions

The analysis made it possible to establish the relationship between heat and mass transfer, chemical reactions and phase transformations in corium core traps during severe accidents at nuclear power plants. The thermal and phase-kinetic characteristics of the dynamics of the interaction of the high-temperature corium melt with the structural element of the sacrificial material are obtained, indicating the mutual influence of these factors on each other.

Acknowledgment

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A Study of Directional Gamma-Ray Detector Without Shield by Monte Carlo Simulation

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I. Introduction

In 2011, the Fukushima Daiichi Nuclear Power Station (FDNPS) was severely damaged by the massive tsunami caused by the Great East Japan Earthquake. At present, many workers are engaged in decommissioning work at the FDNPS. Depending on the place of work, there are locally existing hotspots with high dose rates. To secure the safety of workers, the technology which can grasp the distribution of the radioactive substances in the working environment in advance is required. Information on the spatial distribution of dose rates will also be required when planning decommissioning operations, such as the demolition of reactor buildings to be implemented in the future.

Measuring the spatial radiation distribution of the working environment using survey meters requires a lot of time and effort. Long-time work increases the exposure risk of workers. Also, locally existing hotspots may be overlooked in survey meter measurements. A gamma-ray imager is one of technique that visualizing the distribution of radioactive substances. The gamma-ray imager can be broadly classified into pinhole cameras and Compton cameras. These gamma-ray imager experiments have already been conducted in the field of FDNPS and Fukushima with some results [1], [2], [3].

Since the pinhole camera has a feature that the direction of the radiation source can be determined only by one event, the image reconstruction is simple. However, the total weight is heavy because the entire detector must be covered with a heavy shield (e.g., lead, tungsten). There is also a problem that the collimator limits the field of view (FOV). On the other hand, since the Compton camera does not need a shield in principle, it can be reduced in size and weight. Therefore, our group has developed a system to measure the radiation distribution remotely using a compact Compton camera and has conducted surveys in the FDNPS reactor building [4]. A Compton camera cannot determine the direction of the radiation source with a single event. Therefore, we must overlay multiple Compton cones to determine the direction of the radiation source. Due to this characteristic, the Compton imaging method generates many ghosts of cone traces, which reduce the signal-to-noise (S/N) ratio.

We propose a new gamma-ray imager that combines the advantages of both a pinhole camera and a Compton camera. This feature is achieved by arranging a plurality of directional gamma-ray detectors that do not use shields. The directional gamma-ray detector operates as a Compton camera, sensitive only to small-angle scattering. The new gamma imager discussed in this paper works like a pinhole camera by combining multiple directional gamma-ray detectors. In this paper, the principle verification of the directional gamma-ray detector, which is the fundamental component of the new gamma-ray imager, was carried out using the Monte Carlo simulation.

II. Principle verification using Geant 4 Monte Carlo simulation

A. Concept of the new gamma-ray imager

Fig. 1 shows a conceptual diagram of a pinhole camera (a), a Compton camera (b), and the new gamma-ray imager (c). A pinhole camera comprises a position-sensitive detector inside the shield. The position of the reacted pixel uniquely determines the direction of the radiation source. Therefore, image reconstruction is simple, and there is a possibility of a quantitative measurement of source intensity. However, since a shield is required, the total weight becomes heavy. Therefore, it is difficult to carry and not suitable for measurement by remote operation. A Compton camera is composed of two position-sensitive detectors, namely, a scatterer and an absorber. It uses the kinematics of Compton scattering to estimate the direction

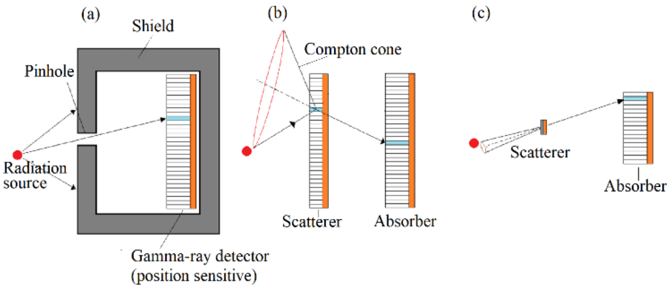

Figure 1. Conceptual diagram (a)Pinhole camera (b)Compton camera (c)New gamma-ray imager

of a radiation source. By taking coincidence with two detectors, and obtain energy information, there is no need to use a shield. Therefore, it is very lightweight and can be mounted on a robot for remote measurement. However, multiple Compton cones must be drawn to determine the direction of the radiation source, which causes the S/N ratio to decrease.

The new gamma-ray imager works like a Compton camera that uses only Compton scattering with a small angle. A plurality of absorbers shares a single scatterer, and the pair of scatterer and absorber constitute a directional gamma-ray detector. The directional gamma-ray detectors do not require shields by taking coincidence and energy information, the same as Compton camera. Therefore, the new detector becomes light and compact, and it can be mounted on robots or drones to measure the radiation distribution remotely. By detecting only small-angle Compton scattering events, it is possible to determine the direction of the radiation source with a single event. Because the source direction can be determined in one event, an image with a higher S/N ratio than that of the Compton camera might be obtained. By arranging a plurality of Fig. 1 (c), it is possible to construct a detector having a wider FOV and detection efficiency. Fig. 2 shows the structure of a single directional gamma-ray detector and the gamma-ray imager as a whole.

Energy threshold set in the scatterer and the absorber distinguish small-angle Compton scattering. The scattering angle of Compton scattering θ and the energy of scattered gamma rays E'_γ adhere to the following equation derived based on Compton kinematics.

$$E'_\gamma = \frac{E_\gamma}{1 + \left({E_\gamma}/{m_e c^2} \right)(1 - \cos\theta)} \qquad (1)$$

Here, E_γ is the energy of incident gamma rays, m_e is the mass of electron, and c is the speed of light. The energy of incident gamma rays E_γ and E'_γ determines the energy of the scattered electrons E'_e as follows.

$$E'_e = E_\gamma - E'_\gamma \qquad (2)$$

When the scattering angle θ_{th} to be cut-off by the scatterer and E_γ are determined, the energy threshold values to be set for

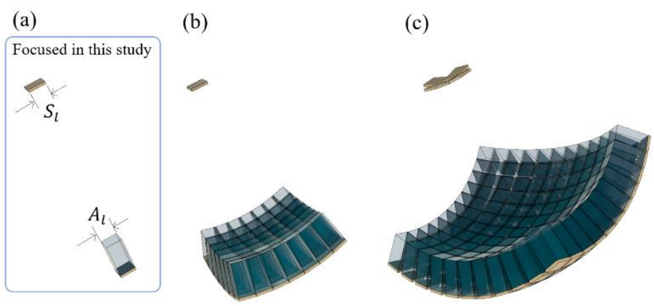

Figure 2. (a) Directional gamma-ray detector (b) Gamma-ray imager consist of multiple directional gamma-ray detectors (c) The FOV can be widened by arranging a plurality of (b).

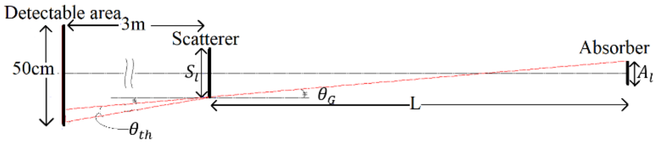

Figure 3. Relationship between the geometry of the directional gamma-ray detector and the detectable area.

the scatterer and the absorber can be calculated. Since radiation from 137 Cs is dominant in the FDNPS working environment, the design assumes 662 keV as E_γ.

B. The basic design of the directional gamma-ray detector

In this study, we investigate a directional gamma-ray detector with a detectable area of 50 cm at 3 m away from the scatterer. The size of the detectable area determines the spatial resolution of the gamma-ray imager. The geometry of the directional gamma-ray detector and the threshold angle θ_{th} determines the detectable area. Fig. 3 shows the relationship between the geometry, the angle threshold θ_{th}, and the detectable area. S_l indicates the size of the scatterer, A_l indicates the size of the absorber, and L indicates the distance between the scatterer and the absorber. θ_G is the angle determined by S_l, A_l, and L. The following conditions must be satisfied in order to obtain a detective area of 50 cm or less at 3 m ahead.

$$S_l + 2\{3\ \mathrm{m} \cdot \tan(\theta_{th} + \theta_G)\} \leq 0.5\ \mathrm{m} \qquad (3)$$

The following equation defines θ_G.

$$\theta_G = \tan^{-1}\left(\frac{S_l + A_l}{2L}\right) \qquad (4)$$

Table 1 shows an example in which specific values are substituted. The larger S_l and A_l, and shorter L improve detection efficiency. However, in this case, in order to obtain a desired detectable area, the scattering angle θ_{th} cut-off by the scatterer must be less than about 2°. In such a small Compton scattering angle, the energy given to scattered electrons is also small as several hundred eV. It is difficult to detect such low energy gamma rays at room temperature. However, if an area having low detection efficiency is excluded from the detectable area, the conditions in Table 1 may be relaxed. The detection efficiency is lower at the edge of the detectable area shown in Fig. 3 than at the center. For example, when the scatterer is irradiated from the center of the detectable area, most of the gamma rays scattered below the θ_{th} can reach the absorber. On the other hand, the edge of the detectable area, even if it is scattered below the θ_{th}, only a part of it can reach the absorber.

TABLE I. Conditions to keep the detectable area within 50 cm

S_l	A_l	L	θ_{th}
10mm	10mm	150mm	0.85°
10mm	7mm	150mm	1.43°
5mm	6mm	100mm	1.57°

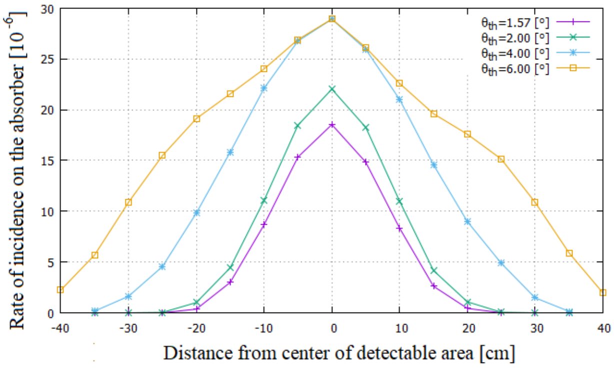

Figure 4. Rate of incidence on the absorber after small-angle Compton scattering. Number of incident gamma rays:10^8, $E_\gamma = 662keV$, Si Thickness:$1000\mu m$, $S_l = 5mm$, $A_l = 6mm$, $L = 100mm$.

The detection efficiency at each point in the detectable area was examined using the Geant4 [5] Monte Carlo simulation. A radiation source is placed at multiple locations on the detectable area, and 662keV gamma rays were irradiated on the scatterer from each point. We investigated the ratio of the gamma rays that reached the absorber after deposit energy below the energy threshold set in the scatterer. The results are shown in Fig. 4.

It can be seen that the detection efficiency decreases as the distance from the center of the detectable area increases. It can also be seen that as θ_{th} increases, the detectable area expands and the peak value increases. However, when θ_{th} becomes larger than a certain value, the peak value does not increase further. This is related to the geometry of the detector. While θ_{th} is smaller than θ_G defined by equation (4), the number of gamma rays reaching the absorber increases as θ_{th} increases. However, when θ_{th} becomes larger than θ_G, not all of the scattered gamma rays are incident on the absorber, and the peak value does not increase further. In the condition of Fig. 4, θ_G is 3.1 °. When θ_{th} exceeds 3.1 °, the peak value does not increase further. Therefore, the maximum detection efficiency is limited by θ_G. Here, the effective detectable area is defined as the full width at tenth maximum (FWTM) when the graph is fitted with a Gaussian distribution. The effective detectable area becomes 56 cm when $S_l = 5$mm,$A_l = 6$mm, $L = 100$mm, and $\theta_{th} = 4°$.

With the geometry described above, it was found that an effective detectable area of 56 cm at 3 m ahead from scatterer was obtained at $\theta_{th} = 4°$. In this case, the energy obtained by the scattered electrons is about 2.08 keV. The scatterer must be able to detect such small energy. When a scintillator is used as the scatterer, such a small signal is buried in noise and cannot be detected. It is desirable to use a semiconductor detector for the scatterer. Among semiconductor materials, Si has a small atomic number and a small photoabsorption cross-section. Moreover, since the Compton scattering cross-section is relatively large and the Doppler broadening effect ([6],[7]) is small, it is suitable for a scatterer.

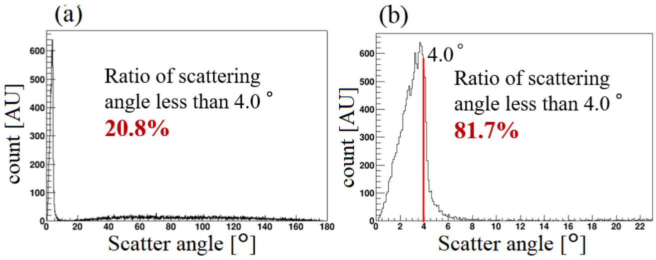

Figure 5. Scattering angle distribution when silicon having a thickness of $1000\mu m$ used as the scatterer. (a) Set an energy threshold of 2.08 keV for the scatterer. (b) A 2.08 keV energy threshold is set for the scatterer and a 600 keV energy threshold for the absorber.

When the energy threshold chooses the small-angle Compton scattering, it must also be considered that the effect of the large-angle Compton scattering is included. Scattered electrons produced by large-angle Compton scattering may escape from the scatterer after giving part of their energy to the scatterer. If the energy deposited in the scatterer before escaping is smaller than the energy threshold, it cannot be distinguished from a true small-angle scattering event. These noises cause the S/N ratio to decrease. The Compton scattering angle distribution was calculated for events with energy deposits below 2.08 keV. The results are shown in Fig. 5. As a result, scattering of 4° or less was about 20% of the whole, and the remaining 80% was large-angle Compton scattering (Fig. 5(a)). Fig 5.(b) shows that the energy threshold is set not only for the scatterer but also for the absorber. By setting an energy threshold of 600 keV in the absorber, scattered gamma rays with Compton scattering of about 23 ° or more can be eliminated. As a result of setting the energy threshold for both scatterers and absorbers, the selectivity for Compton scattering events of 4 ° or less was over 80%.

III. CONCLUSION

We investigated the feasibility of a directional gamma-ray detector without a shield using the Geant 4 Monte Carlo simulation. It was shown that more than 80% of Compton scattering of 4 ° or less could be selected by providing an energy threshold for each of a scatterer and an absorber. It was also found that a detectable area of 56 cm can be obtained 3 m ahead in the case of $S_l = 5$mm, $A_l = 6$mm, $L = 100$mm, and $\theta_{th} = 4°$. Although the detection efficiency of the directional gamma-ray detector alone is low, an improvement in it can be expected by arranging a plurality of directional gamma-ray detectors. In the future, we plan to conduct experiments to discriminate small scattering angle using an actual Si photodiode. The geometry parameters will also be analyzed in detail and optimized.

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Development of one-dimensional optical fiber type radiation distribution sensing method based on wavelength spectrum unfolding

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I. INTRODUCTION

A significant amount of radionuclides were discharged into the environment as a result of the Fukushima Daiichi Nuclear Power Station (FDNPS) accident which occurred due to the large tsunami caused by the Great East Japan Earthquake on March 2011. Decommissioning operations, including the decontamination inside and outside the reactor buildings of FDNPS, are ongoing. To carry out the decommissioning operations of FDNPS, grasping the position of radioactive hotspots is essential. If the position of locally existing hotspots is clear, such information is extremely important for the establishment of the countermeasures for the reduction of external dose to the workers and the efficient decontamination.

For the radiation distribution measurement inside the FDNPS reactor buildings, dose rate measurement using a handheld survey meter were employed by the Tokyo Electric Power Company Holdings (TEPCO) [1]. However, to carry out the radiation distribution measurement over a wide area such as FDNPS reactor buildings, measurement by survey meter is time-consuming and radiation exposure to the workers during the measurement is a critical issue. Moreover, locally existing hotspots might be overlooked because of the point measurement.

Radiation distribution measurement using the gamma-ray imager such as pinhole camera and Compton camera were demonstrated by several institutes inside the FDNPS reactor buildings [2], [3]. The gamma-ray imager is a powerful device which can measure the radiation distribution remotely. However, radiation imager only can measure the incident direction of gamma-ray and difficult to evaluate the relative intensity of radioactive hotspots. Moreover, gamma-ray imager only can measure the gamma source such as ^{137}Cs, thus unable to measure the distribution of beta-emitting source such as ^{90}Sr (^{90}Y), which is one of the dominant radioactive sources inside FDNPS site.

In this study, we focus on the radiation distribution measurement using plastic scintillating fiber (PSF). PSF is an optical fiber type plastic scintillator which is sensitive to the various radiation type (e.g., gamma, beta, neutron) [4]. PSF can measure the distribution of radioactive material one-dimensionally along the fiber using Time-of-Flight (TOF) method, which measures the incident position of radiation to the PSF from the time difference between two signals which reached both ends of PSF [5], [6]. Taking advantage of this characteristic, PSF has been applied inside the FDNPS site to monitor the leakage of the contaminated water from the large container [7]. However, under high dose rate radiation field such as the FDNPS reactor buildings which exceeds the several mSv/h, pile-up event of timing signal and chance coincidence effect in fiber increases, and timing spectrum no more reflects the incident position of radiation [5]. In order to apply PSF to the FDNPS buildings, an alternative method to measure the radiation distribution under high dose rate radiation field is required.

Here, we propose a novel one-dimensional radiation distribution measurement method using PSF based on the wavelength spectrum unfolding. We focused on the fact that the attenuation length of scintillating light along the PSF depends on the wavelength. By measuring the wavelength spectrum of every transmission distance by the spectrometer in advance as a response function, the initial incident position of radiation to the PSF is expected to be estimated using the spectrum unfolding method. This method can avoid the signal pile-up and chance coincidence effect which occurs in the radiation detector with pulse counting mode under high dose rate field because the scintillating light was integrated by CCD-based spectrometer. We have conducted the feasibility study of the proposing method using the ultraviolet irradiation source and radioactive point source.

II. METHODS

A. Principle

The basic concept of the proposing method is to estimate the incident position of radiation to the PSF using the dependence of attenuation length of scintillation light by the wavelength. As shown in Fig.1, the system comprises only PSF and spectrometer. This system measures the wavelength spectrum of the output light at the edge of PSF by the spectrometer. Here, the attenuation length, which defined as the length along the PSF where the intensity of light with specific wavelength falls to 1/e, depends on the wavelength. Therefore, the shape of the spectrum obtained at the edge of the PSF has the information about the distance of light transmission along the PSF. By measuring the response function in advance, which defined as the wavelength spectrum measured at the PSF edge by the spectrometer with every transmission distance, the spectrum which can obtain when measured a certain radiation distribution can be expressed as the convolution of the response function. In other words, proposing method is to find the combination of radiation intensity of every PSF position, which reproduces an observed spectrum.

B. *Experiments*

The irradiation of ultraviolet (UV) excitation source (emission peak: 370 nm) was used to obtain the response function of every transmission distance; irradiation of radiation source was imitated by the excitation of PSF by UV. Experimental setup of UV irradiation is shown in Fig. 2. The UV excitation source was irradiated every 50 cm from the PSF edge up to 6 m to obtain the response function. The spectrometer which comprises the Peltier-cooling type CCD as the photodetector (Ocean Optics QEPro) was used, which has the characteristic of low dark noise.

Fig. 3 and Fig. 4 shows the obtained response function and analyzed attenuation length, respectively. The blue shades in Fig.4 are the wavelength bins used in this study for analysis. The dependence of the attenuation length by wavelength was confirmed, which showed shorter attenuation length in short wavelength, within the range of wavelength bins.

As a radiation measurement test, 1 MBq ^{90}Sr (^{90}Y) radioactive point source was used. In order to avoid the radiation incidence to the spectrometer, silica fiber (Thorlabs BFH22-365, 0.22 NA, High-OH, Φ365, 5m long) was coupled between the scintillating fiber and spectrometer.

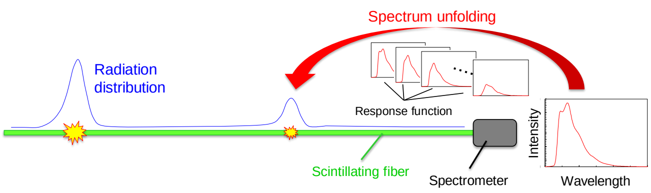

Fig. 1. Schematic of the proposing radiation distribution sensing method using PSF and spectrometer.

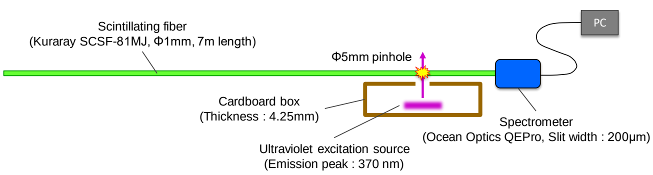

Fig. 2. Experimental setup of UV irradiation to the PSF.

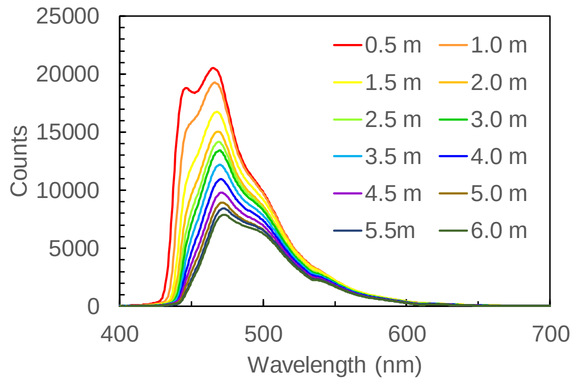

Fig. 3. Response function obtained every 50 cm using UV excitation source.

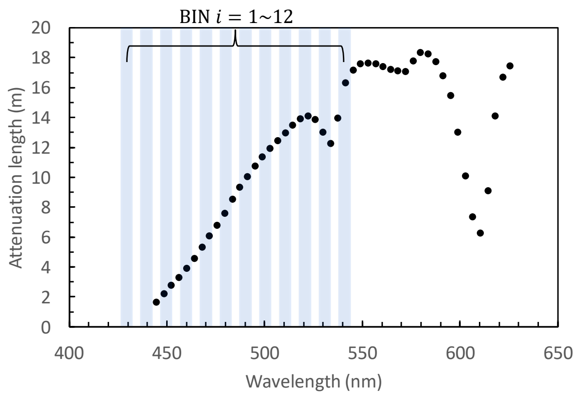

Fig. 4. Wavelength dependency of attenuation length (black dots) and wavelength bins used in this study for analysis (blue shades).

C. *Unfolding method*

The principle of unfolding is to find the combination of response function and radiation distribution, which reproduce an observed spectrum when the certain radiation distribution was measured. The relationship between the response function, radiation distribution and measured wavelength spectrum can be expressed as

$$\begin{pmatrix} \Gamma_{1,1} & \cdots & \Gamma_{1,j} \\ \vdots & \ddots & \vdots \\ \Gamma_{i,1} & \cdots & \Gamma_{i,j} \end{pmatrix} \begin{pmatrix} X_1 \\ \vdots \\ X_j \end{pmatrix} = \begin{pmatrix} A_1 \\ \vdots \\ A_i \end{pmatrix} \qquad (1)$$

Here, a matrix $\Gamma_{i,j}$ standing for the light intensity of the ith wavelength bin of the response function of PSF when the UV irradiated at the PSF position j, X_j $(j = 1, 12)$ is the radiation intensity at the fiber position 50 cm to 6 m every 50 cm which is the solution to be estimated, and A_i $(i = 1, 12)$ is an observed light intensity of ith wavelength bin. Unfolding was carried out

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with a Generalized Reduced Gradient method to optimize the radiation intensity $X_1 \sim X_j$ which minimizes the Δ defined by the following equation,

$$\Delta = \sum_{i=1}^{12} \left(A_{res,i} - A_i\right)^2 \quad (2)$$

where $A_{res,i}$ is a reconstructed light intensity of ith wavelength bin based on the $\Gamma_{i,j}$ and X_j which can be written as

$$A_{res,i} = \sum_{j=1}^{12} \Gamma_{i,j} X_j \quad (3)$$

and A_i is observed light intensity of ith wavelength bin.

The wavelength bins used in this study are shown in Fig. 4. According to the attenuation length also shown in Fig. 4, the wavelength bins were selected widely from the shorter attenuation length to the longer attenuation length.

III. Results and Discussion

A. Multipoint irradiation by UV excitation source

Fig. 5 shows the results of the spectrum unfolding when the UV excitation source was irradiated at multipoint of the PSF. Irradiated positions were reasonably estimated in Fig. 5(a), although it was slightly worse when the UV excitation source was irradiated at 1m and 2m position (Fig. 5(b)). This error may be due to the difference between the wavelength spectrum of multiple point irradiation and the sum of the response function, which may be caused by the slight difference of irradiation geometry (Fig. 6(b)). These spectra should match each other; thus this difference caused the error of estimated irradiation position. Nevertheless, the estimated irradiation positions are consistent with the actual irradiation position. This result suggests that the irradiated position can be estimated reasonably when the irradiated position distribution was pretty simple.

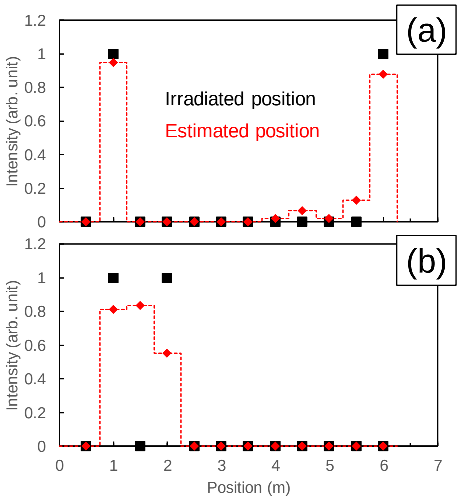

Fig. 5. Comparison of estimated and actual irradiation position when the UV excitation source was irradiated at the multipoint of the PSF.

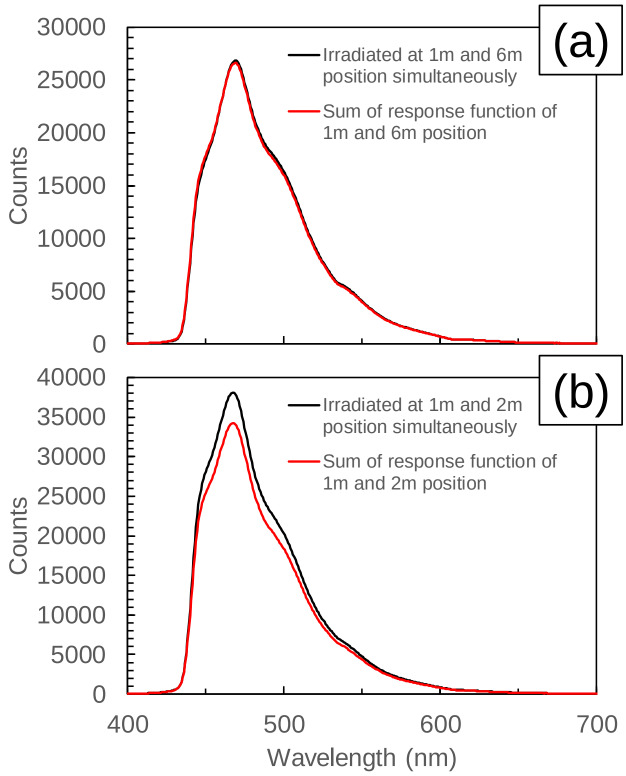

Fig. 6. Comparison of the wavelength spectrum of multiple point irradiation and summed response function.

B. Beta-ray irradiation by ^{90}Sr (^{90}Y) radioactive point source

Fig. 7 shows the measured wavelength spectrum when the 1 MBq ^{90}Sr (^{90}Y) radioactive point source was irradiated at the 1m, 2m, 3m position simultaneously. Integration time was set to 60 sec. Wavelength spectrum originated from the energy depositions by the beta-rays from ^{90}Sr (^{90}Y) to the PSF was clearly observed without any signal amplification (e.g., photo-multiplier) with the spectrometer comprised Peltier-cooling type CCD which has the characteristic of low dark noise.

Fig. 8 shows the results of the spectrum unfolding of the wavelength spectrum shown in Fig. 7 using the response function shown in Fig. 3. Irradiated positions were reasonably estimated. 1m position was estimated clearly; however, 2m and 3m positions were not estimated separately. This result may suggest that the spatial resolution of proposing method is much better at the edge of the PSF near the spectrometer than the other side of the PSF. There may be the two reasons; the first is because the component of long attenuation length become dominant in the response function with long transmission distance, and the difference between the response function is much smaller. For example, the difference between the response function of 1m and 2m shown in Fig. 3 is much larger than the difference between the response function of 2m and 3m, which may indicate that 1m and 2m are much easier to estimate separately. The second is because the signal-to-noise (S/N) ratio became small with the longer transmission distance and uncertainty became larger. It should be noted that the uncertainty factor of Fig. 5 and Fig.8 may not be the same because the spectrum obtained by the UV excitation source has a greater S/N ratio than ^{90}Sr, and the error caused by S/N is negligible.

Using the response function shown in Fig. 3 and the estimated source position and intensity in Fig. 8, the wavelength spectrum was reconstructed in Fig. 7 (red line). Reconstructed wavelength spectrum showed good agreement with measured one, thus the estimated source position shown in Fig. 8 is the result of reasonable unfolding.

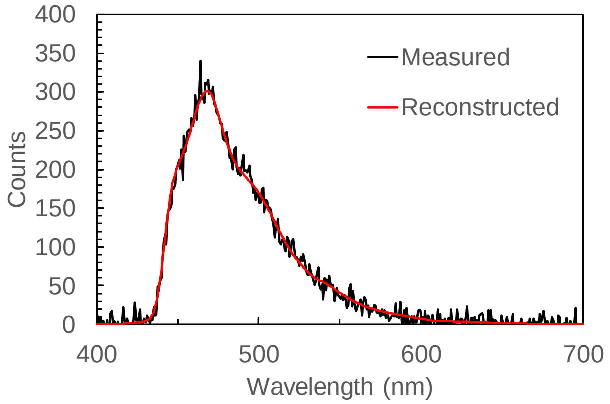

Fig. 7. Wavelength spectrum obtained by ^{90}Sr (^{90}Y) radioactive point source (black) and reconstructed wavelength spectrum using response function and estimated source position in Fig. 8.

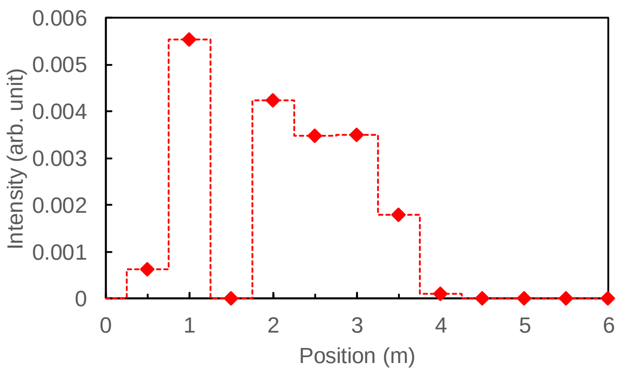

Fig. 8. Estimated source position by the unfolding of wavelength spectrum shown in Fig. 7.

IV. Conclusion

We have developed a novel radiation distribution measurement method using PSF based on the wavelength spectrum unfolding. Proposing method can avoid the signal pile-up and chance coincidence effect, which is typical for the radiation detector with pulse counting mode under high dose rate radiation field. Additionally, unlike the conventional TOF method, proposing method measures the radiation distribution from the wavelength spectrum output only from one side of the PSF, the flexibility of the measurement will improve under the condition with many structures such as inside the FDNPS reactor buildings.

Basic performances of radiation distribution measurement were confirmed using the UV excitation source and the ^{90}Sr (^{90}Y) radioactive point source. The irradiated positions of UV and beta-rays were reasonably estimated using the response functions obtained by the UV excitation source in advance.

The limit of applicability of proposing method is determined by the radiation hardness of PSF. Therefore, we must evaluate the relationship between the total exposure dose to the PSF and the deterioration of attenuation length with every wavelength. Moreover, we must consider applying the optical silica fiber with radiation resistance as the sensor, which is expected to measure the radiation distribution by measuring the Cherenkov light.

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An Algorithm for Reduction in Count Rate Fluctuations, Improved Relative Standard Deviation, Faster Response Time & Spurious Rejection in Nuclear Pulse Counting Systems

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I. Introduction

Radioactive disintegration process being random in nature possesses problem for true count rate estimations while measuring with any instrument. **The process is random both in terms of amplitude as well as time spacing between consecutive events**[1]**.**Hence, the estimation of true count rate becomes difficult more so due to the later. Another problem is for very low count rates, sufficient numbers of pulses have to be acquired to produce statistically correct data which increases the response time at these count rates whereas at high count rates the meter produces data which change at a very fast rate. Hence, hybrid pulse counting methodology has already been implemented[2].

These pulses can be further filtered for suppression of fluctuation as well as detection and rejection of spurious pulses up to certain extent. These processed data also can be checked for Chi squared method for its consistency in fluctuation.

II. Instruments And Methods

A. *Detailed Explanation & Implementation of the Algorithm*

The hybrid method of pulse collection employs a fixed pulse (7 pulses) collection method that is time constrained between 12 seconds and 1 second. i.e. at a count rate of 7 Counts per Second (CPS) or more, the algorithm sticks to constant time interval of 1 second pulse collection routine to estimate the count rate. At a count rate less than this, the algorithm seeks for collection of 7 pulses within the time window of 12 seconds with 1 second increments. Whenever 7 pulses are acquired the CPS is estimated. Under the circumstances when 7 counts could not be collected in even within 12 seconds, the CPS calculation is done with the available counts collected.

Two buffers work in conjunction to analyze and process the pulse counts. One is a **buffer-of-three** (major function of which is spurious checking) whereas the other one is a buffer-of-sixteen (used to output the count rate.)

B. *Processing in Buffer-of-three*

SD (σ): Standard Deviation which is calculated as the square root of the mean value of the data set under consideration. i.e. for a data set {x1,x2,x3…xn) the SD is $\sqrt{(x1 + x2 + x3 + \cdots + xn)/n}$. …**def [1]**

RSD: Relative Standard Deviation= $\sqrt{counts}$ / counts

Note: This definition for SD & RSD holds only for a Poisson or Gaussian distribution, not for any other type [3].

The two buffers are initialized with all zero values stored. The **buffer-of-three** takes three consecutive input valuesprovided by the detector as per the above mentioned methodology. If the latest sample has a value of less than 9, then it is directly passed on to the buffer-of-sixteen, else the Standard Deviation (SD or σ)of the three samples is calculated by taking the square root of the mean value. Each of the three samples is compared for +/- 2σdeviation from the mean value of the three samples. Average value of the set of sample(s), qualifying the test is passed on to the **buffer-of-sixteen**.

Following FIFO logic, the earliest sample is pushed out and latest value is added to the **buffer-of-three** along with two previous values. In this way spurious samples (samples which fail the test) can be detected and discarded from further calculations. Averaging the qualified samples reduces the degree of fluctuation in raw data. The 9 CPS count rate limit is chosen to improve the response time as + σ interval extends up to 9 CPS count rate starting from 0, which would have unnecessarily increased the response time.

C. *Processing in Buffer-of-sixteen*

Input for **buffer-of-sixteen** is received from the **buffer-of-three**. The SD of the data set is calculated as per **def [1].**The explanation for choosing this method to find out the SD is that the fluctuations of the number of pulses from the mean during any particular time constant interval behave in a random way. Therefore, the standard deviation expressed by a square root of the mean count rate is the appropriate measure of the mean count rate error.

All the sixteen samples are checked for **+/-3σ** deviation from the mean. **+/-3σ** deviation is considered to increase the confidence probability at low count rates for more reliability & accuracy. If all the samples are within**+/- 3σ** variation, then the average of these sixteen samples is calculated and the corresponding Radiation value in Engineering Unit is displayed, else the number of samples is reduced by one and then with the latest fifteen samples, above procedure is followed. This process continues until the sample size becomes two in case of failure of the test in stepwise manner. If the test fails for even two samples, then the latest CPS is taken for calculation. In case the algorithm sees a count rate of less than 9 CPS from the beginning itself, only updated values in the **buffer-of-sixteen** are used in the above procedure (and not the zero values which are stored in the **buffer-of-sixteen** during initialization process) till all the sixteen samples are updated. After this the normal sequence follows.

If the latest entry in **buffer-of-three** is more than or equal to 9, then after required calculations in the **buffer-of-three**, the data is pushed into **buffer-of-sixteen**. The SD is calculated using all the 16 values in the **buffer-of-sixteen**. The sixteen samples are checked for +/-2σ variation from the mean value. If all the samples lie inside +/-2**σ** variation, then the average of these sixteen samples is calculated and the corresponding Radiation value in Engineering Unit is displayed, else the number of samples is reduced by one and the above procedure is followed. This process continues until the sample size becomes two in case of failure of the test for each sample size. If the test fails for even two samples, then the latest CPS is taken for calculation.

D. *Chi-squared Test*

After these calculations, using FIFO logic all the samples are moved down the buffer-of-sixteen, the earliest sample being flushed out and making thebuffer-of-sixteenready to accept a fresh sample from **buffer-of-three**. To check out whether the fluctuation pattern is consistent with the expected Statistical fluctuation, chi squared test is performed on all the samples accumulated in the buffer-of-sixteen every time it is updated. From extended chi squared table, the term 'p' is determined corresponding to the statistical degrees of freedom where 'p' may be defined as the probability that a random sample from a true Poisson distribution would have a larger value of chi squared than the specific value showed in the table. This 'p' should be equal to 0.5 for a healthy counting system following Gaussian Statistics. As the value of 'p' drops, it corresponds to more fluctuations whereas larger value of 'p' refers to less fluctuation. As this algorithm aims for reduction in fluctuations, the 'p' value should be equal to or greater than 0.5 [4].Failure of this condition means steady state has not been attended.

The significance of this analysis is evident in the cases where even as all 16 samples are taken for calculation after +/-2sigma scrutiny, there are chances when one can find an unacceptable fluctuation, which may lead to two conclusions. First, the unit is seeing a gradual change in the steady state. Second, if this situation continues for a very long period of time it may imply some fault in the counting system including the software being corrupted.

At a count rate of 7 CPS or more, the algorithm sticks to constant time interval of 1 second pulse collection routine to estimate the count rate. Under the circumstances when 7 counts could not be collected in even within 12 seconds, the CPS calculation is done with the gathered counts for this period of time. Thus the algorithm seeks for a minimum fixed no. of pulses (7 in this case) when the count rate is low whereas it collects pulses for fixed interval of time (1 second) at high count rates.

III. RESULTS, ANALYSIS AND DISCUSSION

A. *Response to step-change in Radiation Rate*

Two sets of twenty five samples (count rates in CPS) each are obtained from a raw nuclear pulse counting system in two different steady states. Assuming a sudden transition from the first state to the second state after twenty-fifth sample, the resulting O/P of the proposed algorithm is obtained. The same data is fed to floating mean algorithm with sixteen samples averaging scheme. Up to first fifteen samples no averaging is done after which a floating mean of previous sixteen samples are produced as O/P.

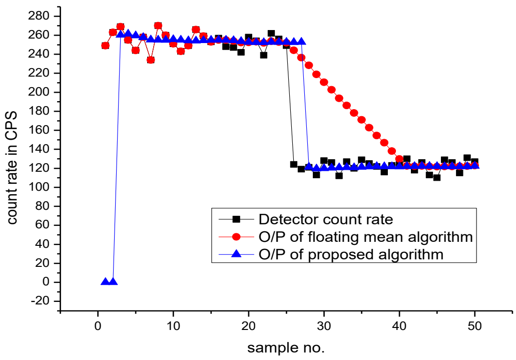

Figure 1. Response to Step-change in Rdiation Rate

TABLE I. ANALYSIS OF ACTUAL DETECTOR DATA

Sample No.	Detector pulse count In CPS	O/P of floating mean algorithm	O/P of proposed algorithm
1	249	249	0
2	263	263	0
3	269	269	260.33
4	255	255	261.33
5	244	244	259.55
6	258	258	257.75
7	234	234	255.26
8	270	270	255.05
9	260	260	255
10	251	251	255.66
11	243	243	255.18
12	249	249	254.43
13	266	266	254.27
14	259	259	254.58
15	253	253	254.95
16	257	255	255.04
17	248	254.9375	254.88
18	247	253.9375	254.62
19	242	252.25	253.7
20	258	252.4375	252.87
21	254	253.0625	252.58
22	239	251.875	252.45
23	262	253.625	252.85
24	256	252.75	252.75
25	249	252.0625	252.81
26	124	244.125	252.52
27	119	236.375	252.79
28	121	228.375	121.33
29	113	218.8125	119.5
30	128	210.625	119.88
31	126	202.6875	120.5
32	112	193.625	120.8
33	127	186.0625	120.94
34	120	178.125	120.76
35	129	171.0625	121.33
36	125	162.75	121.7
37	122	154.5	122.06
38	116	146.8125	121.96
39	123	138.125	121.83
40	124	129.875	121.77
41	130	122.4375	122.04
42	118	122.0625	122.17
43	126	122.5	122.33
44	113	122	122.18
45	110	121.8125	122.1
46	129	121.875	121.89
47	126	121.875	121.85
48	115	122.0625	121.93
49	131	122.3125	122
50	127	122.75	122.37

TABLE II. ANALYSIS OF REM MONITOR DATA

Sample no.	Input CPS	Processed CPS	Remark
1	0	0	
2	0	0	
3	0.24	0.24	Response time: 1 sample (=12 sec), 0.1 mrem/h
4	0.24	0.24	
5	0.24	0.24	
6	0.24	0.24	
7	0.24	0.24	
8	0.24	0.24	
9	2.42	2.42	Response time: 1 sample (=3 sec) 1 mrem/h
10	2.42	2.42	
11	2.42	2.42	
12	2.42	2.42	
13	2.42	2.42	
14	2.42	2.42	
15	24.2	2.42	
16	24.2	24.2	Response time: 2 samples (=2 sec) 10 mrem/h
17	24.2	24.2	
18	24.2	24.2	
19	24.2	24.2	
20	24.2	24.2	
21	242	24.2	
22	242	24.2	
23	242	242	Response time: 3 samples (=3 sec), 100 mrem/h
24	242	242	
25	242	242	
26	242	242	
27	2420	242	
28	2420	242	
29	2420	2420	Response time: 3 samples (=3 sec), 1000 mrem/h
30	2420	2420	
31	2420	2420	
32	2420	2420	
33	24200	2420	
34	24200	2420	
35	24200	24200	Response time: 3 samples (=3 sec), 10000 mrem/h
36	24200	24200	
37	24200	24200	
38	24200	24200	

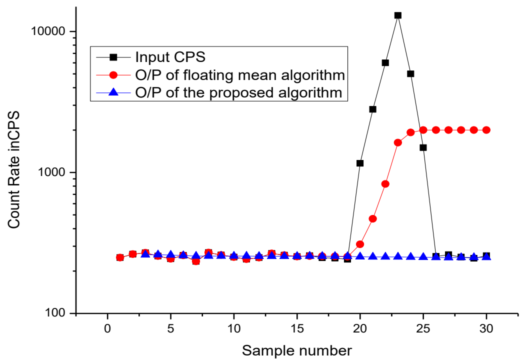

Figure 2. Limited Spurious rejection response

The graph shows quicker response time of the proposed algorithm in comparison with the floating mean algorithm. If number of samples is reduced in the floating mean algorithm to improve the response time, it will adversely affect the RSD factor at low count rates. Hence the proposed algorithm has faster response time at the same time required statistical accuracy. The smoothness of the blue line in the graph shows the fluctuation suppression capability of the algorithm.

The noise rejection capability of the algorithm is described as below. Supposing the previous data set, after twentieth sample, there is an HV induced noise burst for a brief time after which it subsides. The following table shows response of both types of algorithms under comparison, to this event.

B. Case Study of a Neutron REM Monitor

The REM counter under study has a sensitivity of 2.42 CPS/mrem/h to thermal neutrons for the moderator and detector assembly used in it. The following table gives the Calculated Response Time in various ranges. After achieving stable input over all the 16 samples, the following is the table for RSD at various ranges

TABLE III. CO-EFFICIENT OF VARIATION & RSD

Range (value in mrem/h)	Total counts collected	Co-efficient of Variation =SD/Mean	RSD = (SD / Mean) x 100	Time required to achieve this RSD
0.1	46	0.1475	14.75	192 seconds
1	116.16	0.107	10.7	48 seconds
10	387.2	0.0508	5.08	16 seconds
100	3872	0.0160	1.6	16 seconds
1000	38720	0.0050	0.5	16 seconds
10000	387200	0.0016	0.16	16 seconds

IV. CONCLUSION

A hybrid pulse counting algorithm has been designed, developed and proposed. A computer program in "C" Language was developed to simulate the algorithm with satisfactory results. The corresponding algorithm was implemented in embedded system and tested in Radiation Monitors. The results were satisfactory with reduction in count rate fluctuations, improved relative standard deviation, faster response time & spurious rejection. This algorithm is applicable only in case of Poisson distribution or Gaussian distribution only & nuclear disintegration events are mostly governed by these two types of distribution.

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A Comparative Analysis of Regulatory Provisions for Environmental Safety through the Lifecycle of Nuclear Power Stations in the United Kingdom and South Africa

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I. INTRODUCTION

This paper explores the governance, institutional and legal framework within which decisions are made as well as the performance of regulatory bodies within these systems during the lifecycle of the nuclear power station in the United Kingdom (UK) and South Africa (SA). The paper also addresses the issue of high-level nuclear waste (HLW) as an issue that cannot be ignored any longer. The presence of HLW is a fact that will not disappear, whatever happens to the UK or SA nuclear industry. Currently, nuclear stockpiles are growing in both countries, the issue of how to deal with nuclear waste as well as continuous and subsequent legal and regulatory issues are high on the agenda in the UK and SA. It is also important to state from the outset that this paper is concerned with the civil use of nuclear power, whereas, military, reprocessing, research and other uses are outside the remit of this paper. The UK and SA have both expressed interest in the expansion of their nuclear power stations. It is important to note that both countries have not built any new nuclear stations since 1995 and 1985 respectively due to financing and public opposition [1], [2]. The lack of experience has led to the loss of technological capabilities with regards to the construction of nuclear power stations [3]. Consequently, both countries are considering foreign vendors for their nuclear installations. In this context, the importance of effective environmental and safety regulation is imperative, and some legal issues need to be addressed not only in the perspective of the nuclear life cycle but also in the wider context of the management and disposal of nuclear waste.

II. REGULATORY ISSUES

Radioactive materials which are produced during nuclear energy activity generate ionizing radiation [4]. Importantly, radioactive materials pose considerable risks to public health and biodiversity if accidently released into the environment. Additionally, these risks may remain present in the environment for a long time due to the time it takes for radioactive materials to decay to safe levels for humans and other species. Accidents such as Chernobyl in Ukraine (Former Soviet Union in 1986 [5] and the Fukushima Daiichi in Japan in 2011 [6] have demonstrated tremendous risks that are posed by the release of harmful ionizing radiation. The risks include for example, human health effects, social disruption, economic impacts and environmental contamination. Activities such as installation, operation, decommissioning and HLW disposal therefore require extensive legal and regulatory provision to keep the level of risk posed to the public and the environment within acceptable limits. An effective regulatory regime is therefore vital.

Most of the law in this area is statutory, and as such, it articulates the mainstream of political consent as formal standards. [7]. Policy issues and other developments provide the context for discussions of the legal and regulatory regime. In this context, a comparative legal research combined with a sociolegal approach consisting of doctrinal analysis of regulatory structures with policy as context has been used.

III. COUNTRY SELECTION

The UK has been selected for comparison because it is a country with an extensive and well-developed history of nuclear energy utilization and private ownership of the industry. The UK launched its nuclear power program under state ownership and ultimately privatized the industry as part of its wider electricity liberalization strategy in 1997 [8]. Liberalizing was in part intended to replace the command and control structures that existed at the time. Further, the government argued that the liberalization would help reduce the cost of electricity and that consumers would have a choice of their supplier. Nevertheless, the UK currently generates 21% of electricity from 15 nuclear power reactors which are privately owned and managed by Électricité de France (EDF) [9]. Although the UK is embarking on an ambitious program to install 6 new nuclear power stations by 2050, the trajectory of the nuclear power sector is in freefall [10]. In contrast, SA is the only African country with nuclear power technology, and it is an emerging nuclear country with state ownership of the nuclear sector. SA currently has one nuclear power station with two rectors, which generates 5% of its electricity. The Integrated Resource Plan 2019 supports a 20-year life extension for the Koeberg nuclear power station to 2044 and a delayed new build program of 8 new reactors after 2030 [11].

Although the UK and SA differ in many respects, for example, they have distinct geopolitical, government, economics, culture and energy policies, but they also have several similarities which make them suitable for comparison. Some of the similarities are that; (1) both countries originally developed nuclear power for military use (2) both countries have lost their technological capabilities in installation of nuclear power stations and exploring foreign vendors to fulfil this role, (3) both countries are seeking a solution for HLW (4) both countries have faced bureaucratic procedures (5) both countries have had change of governments which can contribute to regulatory challenges. With these issues in mind, comparison of these two countries may hold important lessons for the countries that are aspiring to undertake similar nuclear energy projects. The subsequent sections explore the governance and the regulatory provisions for environmental safety in which the nuclear industry operates.

A. *The UK governance and legislation and regulatory regime*

The UK operates as a constitutional monarchy, with parliamentary of governance and it is a state that implements a dual approach to international law and the supremacy of European Union (EU) law until Brexit [12]. The UK operates a devolved energy sector, whereby, the UK government has the competence for the installation of nuclear power stations, however, the Scottish parliament has competence over the land use and environmental regulations, and an energy policy which is opposed to the installation of new nuclear power stations. However, the Scottish Government's policy is open to an option of small modular reactors [13]. The complex system and the legal powers accorded to the devolved government of Scotland presents a significant conflict and obstacles to the UK government energy policy implementation. In this context, the Scottish government have the power to block the installation of new nuclear build under the planning and environmental law.

The basis of the licensing regime is the Nuclear Installations Act 1965, and the regulatory authority is an Independent Office for Nuclear Regulations (ONR), that has the legal responsibility for the regulation of nuclear safety. The UK's approach to nuclear safety is not prescriptive, it has a "one-step" licensing approach of the nuclear new build. According to the licensing regime, the operator is required to assess and manage safety proactively through the lifecycle of nuclear power stations including waste management, storage and disposal. However, the International Atomic Energy Agency (IAEA) recommends a step by step licensing process, in part because it allows rigorous regulatory oversight of the process [14]. Part of the challenge of disposal in a GDF is working out how the nuclear licensing system should apply as the risks are quite different to an operational nuclear power station. The government and ONR has had issues with regards to this matter and are in the process of devising new regulations and continue to work on these issues [15].

The radioactive substances and disposal of radioactive substances is provided by the Radioactive Substances Act 1993. The UK government's policy with regards to nuclear waste management has gone through dramatic and controversial changes. The UK has found it difficult to implement consistent policies for HLW waste management. The UK's independent, Committee on Radioactive Waste Management (CoRWM) has recommended that government should consider Geological Facilities for the storage of HLW which is currently being stored on nuclear energy sites. The company responsible for finding a solution for HLW disposal is Nuclear Decommissioning Authority (NDA). The key issues are the and ensuring that waste is packaged safely, and for ultimate disposal. part of the challenge of disposal in a GD is working out how the nuclear licensing system should apply as the risks are quite different to an operational nuclear reactor. This paper examines several legal issues that need to be addressed not only in the context of environmental protection, but the wider key issue of management of HLW nuclear waste.

B. *SA govenance and legislative and regulatory regime*

SA is a parliamentary Republic run by a president who is both head of State and the head of government and decisions for energy power are centralized. SA is an emerging nuclear country where an estimated 11% of the people do not have access to electricity and 4% rely on illegal access [16]. SA acquired nuclear power technology was motivated by the weapons in the 1940s, but in the 1980's the SA government explored the possibility of using nuclear energy [17].

The basis for licensing in SA is the Nuclear Energy Act 1999 (Act 46 of 1999) which also established the National Nuclear Regulator (NNR). Eskom is the owner and operator of the Koeberg nuclear power station. SA has no codes or guidelines for the national nuclear industry. Therefore, the NNR's approach is non-prescriptive. The NNR relies on vendor codes of conduct to compile necessary NNR requirements and local conditions. The operator is required to submit a safety case to demonstrate compliance with the regulatory requirements under the NNR requirements, and if satisfied the NNR then issues a nuclear license that enforces the safety standards and holds the operator to their commitments. The National Radioactive Waste Disposal Institute Act 2008 (Act 53 of 2008) established the National Radioactive Waste Disposal Institute (NRWDI) 2014, to address radioactive waste management. The policy has been criticized for being weak as it merely contains a checklist of options that future policymakers could select. The policy document fails to address a long-term strategy for a long-term solution for HLW, it is unclear what the government's plans are with regards to HLW. The NRWDI has also been criticized for lacking clear goals and inaction concerning HLW [18]. Currently, HLW is stored in reactor pools on-site, at the Koeberg nuclear power station. Vaalputs was once proposed as interim storage for HLW, however, Vaalputs is 700 km away from Koeberg and concerns surrounding safety issues have been raised due to the long distance that HWL has to travel on the unsafe and accident-prone highways of SA [19]. The Vaalputs Radioactive Waste Facility is currently operated by

Nuclear Energy Corporation (NESCA, owned by the state) for the storage of low-level nuclear waste [20].

IV. Conclusion

The UK and SA seek to expand the capacity of their nuclear power stations, radioactive waste management being one of the central themes in the debate around nuclear new build for both countries. The UK was one of the architects of nuclear energy and a well-developed nuclear industry in the world, however, the trajectory of the industry is shifting from a position of nuclear energy leadership to a declining nuclear industry seeking international vendors for construction and private financing and insurance. Both countries lack technological capabilities because there have not built any nuclear power stations for a long time. The result is that both countries will have to rely on foreign vendors to support the installation, decommissioning and management of radioactive waste. The emphasis of this paper has been to analyze the regulatory, environmental provisions of exploring how nuclear energy regulations function within, across and around national, regional and international frameworks that govern them. The UK and SA have demonstrated a weak legacy of radioactive waste management. Despite the issues highlighted in this paper, both countries are in favor of expanding their nuclear fleet and produce even more radioactive waste and therefore emphasis should be placed on an effective legal and regulatory regime.

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Integration of Fission Product Release Analysis from Reactor Pressure Vessel and Spent Fuel Pool in Modified ART Mod 2

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I. Introduction

In 2011, the accident at Fukushima Daiichi Nuclear Power Plant (1FNPP) affected people and environment worldwide including both countries who use and do not use nuclear energy [1]. Transboundary atmospheric dispersion of fission products becomes an important issue for the countries having chance to receive consequences from severe accidents of a nuclear power plant (NPP) [2].

For ASEAN countries, although, there are no uses of nuclear energy, they realize the importance of safety from use of nuclear power from other countries especially neighbor ASEAN countries [3]. Therefore, ASEAN Network on Nuclear Power Safety Research (ASEAN NPSR) is organized to study nuclear power safety in ASEAN region.

Under ASEAN NPSR framework, transboundary atmospheric dispersion assessment of fission products from severe accidents in a nuclear power plant (NPP) is being performed in order to support the strategic planning of accident management [4]. The scope of this paper focuses on consequences of fission product releases from NPPs around ASEAN countries to ASEAN region.

Meteorological data and source term data are important inputs for the transboundary dispersion assessment. In previous study, meteorological data was obtained from the US National Oceanic and Atmospheric Administration (NOAA) [5] and source term data was obtained from publicly available data of other reactors such as State-of-the-Art Reactor Consequence Analyses (SOARCA) [6].

There are different NPP types around ASEAN region of which characteristics and amount of source term are unique. Therefore, use of publicly available data of other reactors from SOARCA [6] in the transboundary dispersion assessment may not accurately represent source terms of specific NPPs. It is important to develop a tool to determine source terms to be used in the transboundary dispersion research.

Thailand have been using ART Mod 2 of Japan Atomic Energy Agency (JAEA) [7] to analyze the source term since 2012. ART Mod 2 has been modified and validated for the evaluation of fission product release from reactor pressure vessel (RPV) into containment vessel (CV) in 2019 by Thailand Institute of Nuclear Technology (TINT) in cooperation with Chulalongkorn University (CU) [8]. It was also used to fission product releases from Spent Fuel Pool (SFP) into the environment [9].

To enlarge the framework of fission product release analysis and to develop a tool to determine source terms to be used in the transboundary dispersion research, this research integrates the evaluations of fission product releases from the RPV and from the SFP to the CV in Modified ART mod 2 in order to achieve equivalent capability to well-known codes, such as MELCOR 2.1 [10, 11]. As the release into the environment mainly comes from the fission products in the CV, the source term in the CV is the upper boundary to the environmental source term, which is defined as maximum environmental source term in this study. It is noted that there are other possible source terms in actual accidents.

The objective of this study is to integrate evaluations of fission product releases from the RPV and the SFP into the CV in Modified ART Mod 2 and verify the simulation results with MELCOR 2.1 simulation data of fission product releases from a RPV of Unit 3 of the accident at 1FNPP [10], and from a SFP of Mark I Boiling Water Reactor (Mark I BWR) [11] which is the same type as that in the Unit 3 of 1FNPP. Unit 3 of the 1FNPP was selected because of it was suspected to experience melted core in the RPV and fuel damage in the SFP [12]. Cesium hydroxide (CsOH) is selected as the representative source term. This is because CsOH was the majority of the release of cesium (Cs) compounds after NPP accidents [8, 13]. Moreover, Cs compounds affect people and the environment in a long term due to a half-life of 30 years [14].

II. BACKGROUND

A. ASEAN NPSR

ASEAN NPSR is a group of researchers who realize the importance of ensuring safe use of nuclear power in ASEAN region. ASEAN NPSR's goal is to reinforce nuclear power safety in ASEAN region through research and development (R&D), human resource development and regional cooperation in order to support the preparation of the regional strategy for accident management corresponding to the IAEA Safety Standards and lessons learned from the Fukushima Nuclear Accident [3].

B. Deposition Phenomena in Modified ART Mod 2

Modified ART Mod 2 is a code for studying phenomena deposition of fission products in gas form and aerosol form [7]. There are two regions on which fission products can deposit including wall and floor. Fig. 1 shows overview of deposition phenomena of fission products in gas form and aerosol form of Modified ART Mod 2.

Deposition phenomena in gas form include (1) condensation and (2) adsorption on wall. As for aerosol, deposition phenomena include (1) gravitational settling on floor, and (2) Brownian diffusion, (3) diffusiophoresis and (4) thermophoresis on wall. The details of phenomena are shown in Table I.

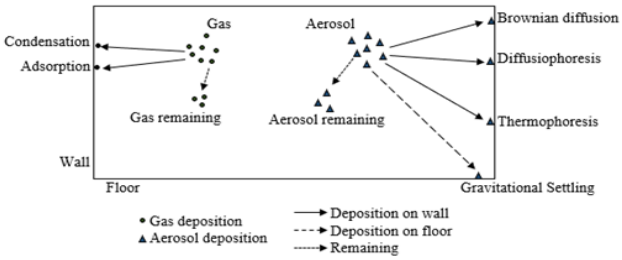

Figure 1. Overview of deposition phenomena of fission products in gas form and aerosol form of Modified ART Mod 2.

TABLE I. DETAILS OF DEPOSITION PHENOMENA OF FISSION PRODUCTS IN GAS FORM AND AEROSOL FORM OF MODIFIED ART MOD 2

Deposition phenomena	Details
Condensation	A deposition phenomenon on wall which occurs from differences between partial pressure and saturated vapor pressure in the system
Adsorption	A deposition phenomenon on wall which occurs from reaction of radionuclides and surface of material in high temperature condition
Gravitational settling	A deposition phenomenon on floor which occurs from effect from gravitational force
Brownian diffusion	A deposition phenomenon on wall which occurs from no direction movement of a particle immersed in a fluid
Diffusiophoresis	A deposition phenomenon on wall which occurs from the flow of the condensing steam and partial pressures of non-condensable gas near the structure surface.
Thermophoresis	A deposition phenomenon on wall which occurs from differences of temperature gradient.

C. MELCOR computer code

MELCOR is an integrated computer code that was designed to study progression of severe accidents of light water reactor NPPs [15, 16]. This code studied severe accidents from thermal-hydraulic response in the reactor coolant system (RCS) to fission product release into environment. For phenomena depositions of fission products, MELCOR uses the similar models as in Modified ART Mod 2 but there are differences of the correlations and conditions in each model.

III. METHODOLOGY

Accident in the RPV and the SFP of Unit 3 of the 1FNPP resulting in CsOH release into the CV was selected as a representative. The calculation is divided to three steps below.

A. Step 1: CsOH Release from the RPV into the CV

Nodalization of the RPV and the SFP to the CV is shown in Fig. 2. It consists of volumes of the RPV and the SFP at the bottom and a volume of the CV on the top. Geometric parameters are shown in Table II. Timeline of the simulation is 42 – 72 hours after the initiation of the accident in Unit 3 of 1FNPP which is the range from core melt to hydrogen (H_2) explosion (108,000 second) [17]. The conditions of the simulation are shown in Table III.

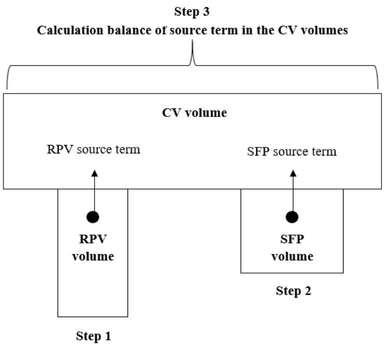

Figure 2. Nodalization of the RPV and the SFP to the CVs in Modified ART Mod 2.

TABLE II. GEOMETRY PARAMETERS OF THE NODALIZATIONS

Volume	Geometry Parameters		
	Height [cm]	*Width [cm]*	*Length [cm]*
RPV	2.10×10^3	5.50×10^2 (Diameter [cm])	
SFP	1.22×10^3	1.20×10^3	1.20×10^3
CV	1.58×10^3	4.60×10^3	4.60×10^3

TABLE III. SIMULATION CONDITION OF CSOH RELEASE FROM THE RPV INTO THE CV

Parameters	Values
Initial activity of source term of CsOH in RPV [18]	2.14×10^{17} Bq (100% of core inventory)
Temperature of source term [19]	1,280 K – 3,000 K
Average wall temperatures of the RPV [17]	950 K
Average wall temperatures of the CV [20]	287 K
Average pressures in the RPV [11]	0.3 MPa
Average pressures in the CV [17]	0.1 MPa

TABLE IV. SIMULATION CONDITION OF CSOH RELEASE FROM THE SFP INTO THE CV

Parameters	Values
Initial activity of source term of CsOH in SFP [18]	3.29×10^{17} Bq (100% of core inventory)
Temperature of source term [19]	1,280 K – 3,000 K
Average wall temperatures of the SFP [11]	450 K
Average wall temperatures of the CV [20]	287 K
Average pressures in the SFP [11]	0.1 MPa
Average pressures in the CV [11]	0.1 MPa

B. *Step 2: CsOH Release from the SFP into the CV*

Timeline of the simulation was in the same range of the CsOH release from the RPV. The conditions of the simulation are shown in Table IV.

C. *Step 3: Balance of CsOH Releases in the CV*

After the CsOH releases from the RPV and the SFP into the CV in Fig. 2 were calculated, the total CsOH releases in the CV were balanced between the two simulations to discuss about the maximum environmental source term.

IV. RESULTS AND DISCUSSIONS

The simulation results of the CsOH release from the RPV and the SFP into the CV help calculate amount of CsOH depositions on wall and on floor in each phenomenon in gas and aerosol forms. To verify Modified ART Mod 2, amount of CsOH releases are compared with MELCOR 2.1 results.

Fig. 3 and Fig. 4 show the results of depositions of the CsOH releases from the RPV and the SFP into the CV in Steps 1 and 2 respectively. It is found CsOH releases in aerosol form deposit on wall due to thermophoresis model of Modified ART Mod 2. Thermophoresis is dominant because effect of high temperature gradient of source term and wall to thermal conductivity increase in the model [8]. While, CsOH gas deposits on wall due to absorption which is phenomenon at high temperature condition of gas form [8]. In observation, there are CsOH releases in aerosol form from the RPV into the CV more than from the SFP because average pressure in the RPV from the initial conditions is higher than average pressure in the SPF. High pressure affects aerosol agglomeration directly [7].

Fig. 5 and Fig. 6 show comparisons of the CsOH releases from the RPV and the SFP between Modified ART Mod 2 and MELCOR 2.1 respectively. In Fig. 5, the total release of CsOH using Modified ART Mod 2 was calculated from total CsOH in gas form and aerosol form in Fig. 3. It was found that the total release of CsOH using Modified ART Mod 2 is slightly overestimated when compared to MELCOR 2.1. The values of the two CsOH releases can be approximated to be 1% of core inventory. While, in Fig. 6, It was found that total releases of CsOH using Modified ART Mod 2 and MELCOR 2.1 [11] have less release than 1% of core inventory.

From the comparison of the CsOH releases in Fig. 5 and Fig. 6, it is found that the CsOH releases using Modified ART Mod 2 are consistent with MELCOR 2.1 results. Although, each phenomenon has slightly different values because of differences of the correlations and conditions in each model, the values of the two CsOH releases are in the same order.

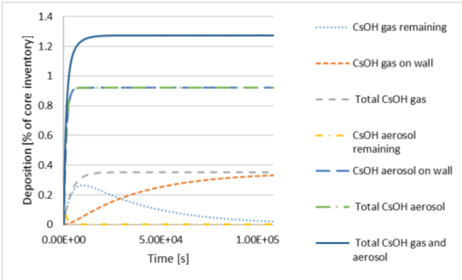

Figure 3. Depositions of the CsOH release from the RPV into the CV using Modified ART Mod 2.

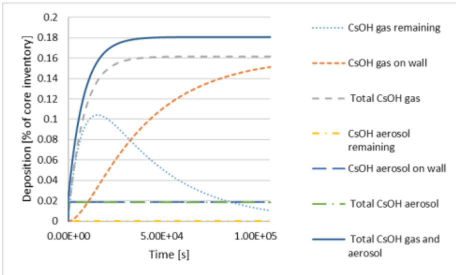

Figure 4. Depositions of the CsOH release from the SFP into the CV using Modified ART Mod 2.

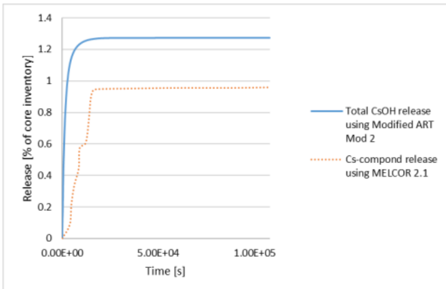

Figure 5. Comparison of the CsOH releases from the RPV between Modified ART Mod 2 and MELCOR 2.1

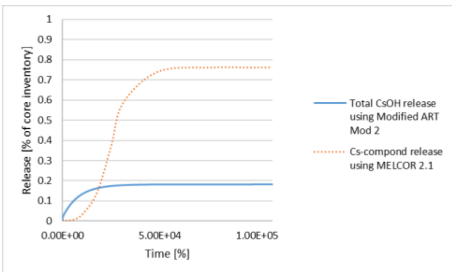

Figure 6. Comparison of the CsOH releases from the SFP between Modified ART Mod 2 and MELCOR 2.1.

TABLE V. COMPARISON OF CS RELEASE FROM MEASUREMENT DATA WITH CALCULATION DATA USING MODIFIED ART MOD 2

Measurement data [19]	**Calculation data using Modified ART Mod 2**		
Cs release during early H2 explosion of Unit 3 [Bq]	*Maximum CsOH releasing from the RPV [Bq]*	*Maximum CsOH releasing from the SFP [Bq]*	*Maximum CsOH releasing from the RPV and the SFP [Bq]*
2.97×10^{14}	3.06×10^{15}	3.65×10^{15}	7.31×10^{15}

The trends of depositions in Modified ART Mod 2 and MELCOR 2.1 [10, 11] are quite consistent because the two codes considered similar deposition phenomena. The calculated release amount may vary though it is still in the same order of magnitude. Therefore, Modified ART Mod 2 can be considered acceptable for the integrated calculation of the release from the RPV and the SFP.

In Step 3, the total CsOH release in the CV is balanced between the releases from the RPV and the SFP to discuss about the maximum environmental source terms. Table V showed the comparison of Cs release during early H_2 explosion of the Unit 3 of the 1FNPP from measurement data [21] with calculation data using Modified ART Mod 2. Maximum source terms in Table V are calculated by multiplying fraction of CsOH release in Fig. 4 and Fig. 6 with amount of core inventory of the RPV and the SFP, respectively. If CsOH releasing from melted fuels in the SFP is taken into account, the released CsOH may increase by two-fold from the contribution of the release from the SFP. From this study, Modified ART Mod 2 helps predict the maximum environmental source terms when releases from the RPV and the SFP occur in the same time.

V. CONCLUSIONS

Modified ART Mod 2 was used to perform integrated evaluations of releases from the RPV and the SFP into the CV and was verified with MELCOR 2.1 results.

Due to the similar deposition phenomena of Modified ART Mod 2 and MELCOR 2.1, there are non-significantly different results. Therefore, Modified ART Mod 2 are acceptable to calculate the release from the RPV and the SFP.

In addition, Modified ART Mod 2 can predict the maximum environmental source terms when releases from the RPV and the SFP occur together.

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Microwave Technology for Personal Monitoring as a Nuclear Security Application

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I. Introduction

One of the most crucial aspects of nuclear security is prevention, or detection and response, to deliberate malicious events involving radioactive materials or directed against facilities or activities where such substances are used [1, 2, 3, 4]. Nuclear security involves monitoring of the personal seeking entrance or exit to any nuclear facility through a security check. It is a common practice in nuclear facilities to have multiple physical barriers for security checking [4, 5]. The trending imaging tools are X-ray, Gamma-ray, and Ultrasound. As the frequency of X-ray and Gamma rays are very high (> 10^{15} Hz), it will cause significant damage to the personal as compared to microwaves (~10^{10} Hz). Ultrasonic application for security checks will not be feasible as it requires a medium to propagate through to image the object [6]. In the nuclear industry, the usual trend is to scan the objects and personal separately for nuclear security. The idea of separate monitoring of personal draws the attention towards the microwave imaging. Microwave imaging can help in locating anything present on the body with no harm. In addition, the dielectric properties of the objects can also be measured. This paper discusses the synthetic aperture radar (SAR) algorithm [7, 8] for locating the shape and size of the test objects. SAR is a qualitative description of the test object which encourages the need to involve quantitative analysis. This paper proposes a quantitative "microwave time-domain algorithm [9] for multilayered media" to identify the relative dielectric properties of the test objects identified in the test region using SAR. Various test objects used in the experiment are papers and diary as important documents, metal plate and CD as threat element, hard disk, and mobile as an electronic instrument, and purse.

II. Measurement Setup

The microwave measurement setup requires a vector network analyzer (VNA), antenna, and display unit as the computer, as shown in figure 1. The measurements are carried out by scanning the test area, which involves an antenna triggered by VNA. The recorded data on VNA is then acquired on PC to perform analysis and imaging. The experimental setup and mannequin as a test area of scanning are shown in figure 2. The microwave measurements are performed using a horn lens antenna operable from 1 GHz-18 GHz and a VNA operable from 5 kHz - 20 GHz. In the experiments, 15 GHz – 18 GHz frequency range has been used for SAR imaging to achieve a significant resolution.

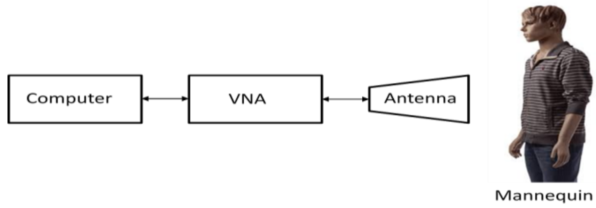

Figure 1. Block Diagram of Microwave Measurement setup.

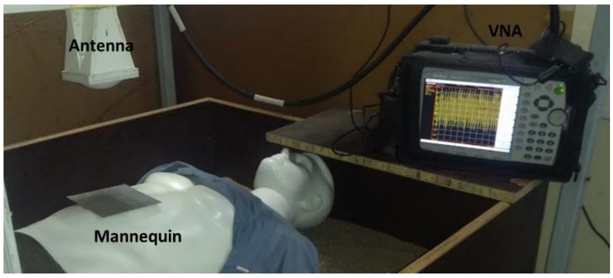

Figure 2. Microwave Measurement setup.

III. Measurement Method

There are two measurement methods have been employed (i) synthetic aperture radar (SAR) algorithm, and (ii) microwave time-domain algorithm. SAR algorithm plots the reflection coefficients of the test area in the distance domain to locate the shape and size of the object. The latter can determine the dielectric properties of the objects identified in the SAR algorithm. The latter method requires additional data, incident reflection coefficients, which can be measured in advance by placing the metal plate at various locations. The incident and reflected data are then transformed into time-domain to observe the peaks. The detailed description can be studied from literature.

The basic equation to measure the dielectric constant ε_r is [9]

$$\frac{\varepsilon_{rk}}{\varepsilon_{r(k-1)}} = \left(\frac{1+|\Gamma_k|}{1-|\Gamma_k|}\right)^2 \tag{1}$$

$$\text{Where, } |\Gamma_k|^2 = \frac{P_{rk}}{P_{ik}} \quad (2)$$

P_i and P_r are the incident and reflected signal by placing a metal plate at the first interface and removing it, Γ_k is the reflection coefficient.

IV. Results

A. SAR

In this section, 2D shape and size of the CD as high dielectric object and Purse as a low dielectric object placed on the mannequin have been identified. Figure 3a shows the Mannequin used in the experiment; figure 3c shows the mannequin with the test object mounted; figure 3b with test objects covered with a shirt. Figure 3d is the reconstruction of the test objects present on the body using the SAR algorithm. The test objects for imaging are CD and Purse. CD is a high dielectric object with high reflectivity, whereas, the purse is a low dielectric object (low reflectivity). This raw reflectivity of the test objects has been observed in figure 3d. The reflectivity has been modified using the SAR algorithm, and the improved image of the test objects is presented in figure 3e.

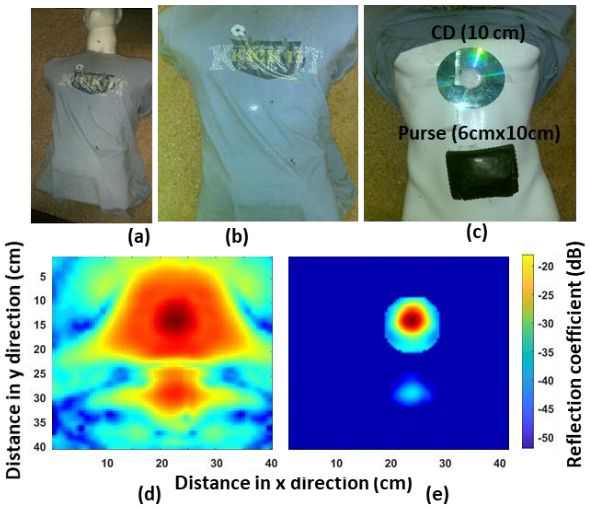

Figure 3. (a) Mannequin, (b) Mannequin with test objects covered, (c) Mannequin with test objects (d) Raw Image of test objects, (e) SAR Algorithm: Plot.

B. Dielectric Measurement

To demonstrate the benefits of SAR imaging, the reflectivity image of the test objects were shown in figure 3. SAR algorithm plots the reflectivity of the test objects with respect to the distance from the source antenna. However, the dielectric nature is still to be discovered.

In this section, the dielectric constant of the test objects has been calculated using the measured reflection coefficient. The dielectric constants are compared for both the situations by covering and uncovering the mannequin with a shirt. The dielectric properties of the Mannequin and surface of the human body have also been measured and compared.

TABLE I. Dielectric Measurement

Dielectric Objects	Dielectric Constant of test objects	
	Uncovered	***Covered***
Metal Object	3061.43000	3540.01000
Hard Disk	87.24390	162.87000
Mobile (back side)	34.42430	44.63100
Mobile (front side)	52.45420	66.50240
Compact Disk	29.90150	30.47380
Papers	2.89419	2.47585
Diary	2.12942	1.79190
Purse	2.01529	2.50592
Human Body	1.37216	1.40842
Mannequin	2.46309	2.39267

V. Conclusion

The results obtained using the SAR algorithm, and the time-domain algorithm was presented in this paper. Various test objects have been located and identified using the combination of both the methods. The motive of implementation of the combination of algorithm, there is no need to perform any separate measurement. The results clearly depict the hidden object on the body. The purpose of nuclear security to detect any object present on the body and has been obtained.

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The Complexities of Layered Obligations in International Nuclear Commerce Peaceful Use Agreements

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I. BACKGROUND

As of September 2019, there are 448 nuclear power plants in operation worldwide, with 53 more currently under construction, producing 14% of the world's consumed energy [1]. An additional 250 reactors in 55 countries generate power for research, training, and the production of medical and industrial isotopes [2].

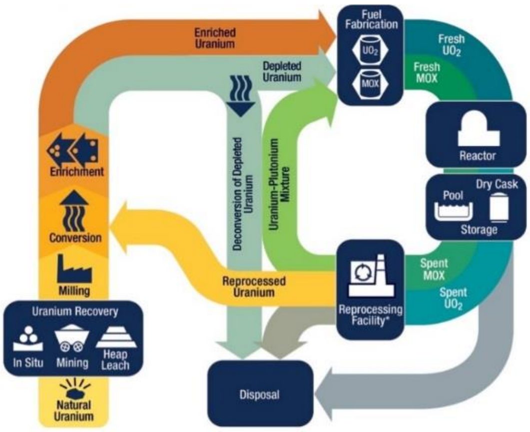

Figure 1. Nuclear Fuel Cycle

While there are differences in the exact fuel type used by some of these reactors, all reactors make use of the nuclear fuel cycle for the production of fuel and disposal of waste (see Figure 1). The nuclear fuel cycle has different steps dependent on the type of reactor being serviced (i.e. a light-water reactor versus a heavy-water reactor versus a fast-breeder reactor).

Servicing the world's nuclear power plants is a global enterprise. Supply chain services range from uranium mining and milling, conversion, and enrichment, to reactor fuel fabrication. The supply chain for fuel frequently involves collaboration between multiple facilities in the fuel cycle. While many countries participate in the nuclear energy industry, only Russia, the United States (US), and China have the complete range of services needed to independently complete the fuel cycle from start to finish. US national policy currently prevents commercial utilities from reprocessing fuel, however the US Department of Energy's (DOE) Savannah River Site regularly reprocesses foreign fuel assemblies into stable waste forms for final disposal. However, there is no economic incentive for commercial facilities to reprocess spent fuel to extract the usable portion in the US when compared to the cost of producing new uranium. Table 1 shows the breakdown of services offered by several nuclear trading partners.

TABLE I. OBLIGATED MATERIALS AND SERVICES PROVIDED BY MAJOR U.S. NUCLEAR TRADING PARTNERS [3]

Fuel Cycle Services	Countries				
	Australia	Canada	EURATOM	Japan	Urenco*
Mine natural U, produce yellowcake	●	●			
Convert yellowcake to UF6		●	●		
Enrich natural U to LEU			●		●
Fabricate fuel rods		●	●		
Operate nuclear power reactors		●	●	●	
Operate research reactors	●	●	●	●	
Reprocess SNF			●	●	
Produce large nuclear plant components				●	
Enrichment technology					●

Many countries, including the US, often go to other countries to seek the most favorable terms for services for conversion, enrichment, and fuel fabrication. For example, a U.S. facility may purchase Australian or Canadian natural uranium. The yellowcake may then be converted to UF_6 in either Canada or Europe, and enriched by technology owned by Urenco in the Netherlands, Germany, United Kingdom (UK), or US before being fabricated into fuel.

In an effort to deter proliferation of nuclear explosives, a framework for peaceful use cooperation between countries engaged in nuclear commerce was developed starting in the 1950s. This framework, formalized in nuclear cooperation agreements between trading partners, requires assurances that nuclear materials as well as associated equipment and technologies be restricted to peaceful uses. These assurances are posted to these items as "obligations" on the part of the receiving country. Obligations require the receiver to track and report to the origin trading partner(s) that the items or material received have been restricted to peaceful uses.

II. PEACEFUL USE OBLIGATIONS

Section 123 of the 1954 Atomic Energy Act of the US outlines the terms for entering into agreements with foreign partners for cooperation in nuclear trade. Nuclear Cooperation Agreements (NCAs, also known as "123 Agreements") are a prerequisite for US exports or imports of nuclear materials, equipment, and technologies. As of July 2019, the US has 23 active NCAs with 48 countries[1], plus the IAEA and Taiwan. [4]

NCAs establish legal requirements for reporting and tracking imported nuclear material, equipment, or technology to ensure compliance with peaceful use commitments. These commitments include:

- Ensuring material, equipment, or technology is used exclusively for peaceful purposes;
- Preservation of IAEA safeguards;
- Commitments to provide physical protection of material, items, and/or technology to avert potential theft or diversion to unauthorized uses;
- Obtaining permission to physically alter or transfer any material acquired through import such as enrichment, reprocessing, or export; and
- Compliance with tracking and reporting requirements specified by the origin trading partner(s) [5]

Nuclear material, equipment, and technology which are subject to the terms of a NCA are also considered obligated to peaceful uses specified by the originating trading partner. Foreign obligations may apply to special nuclear material, source material, nuclear equipment, and nuclear technologies. Because of these restrictions, most domestic fuel cycle facilities have at least some foreign obligated nuclear material and equipment.

A. Receipt of Foreign Obligated Materials

There are two ways that a US facility may acquire foreign obligated nuclear material: (1) receipt of obligated material from other facilities, either international or domestic, with the obligation information documented by the shipper; and (2) adding or creating new obligations by using obligated nuclear processing equipment (e.g., conversion, fuel fabrication, or enrichment technologies) or by use of obligated equipment such as components in power or research reactors.

B. Reporting Requirements of NCAs

US facilities that store, manage, or use foreign material began reporting data to the US government state system of accounting and control (SSAC), the Nuclear Materials Management and Safeguards System (NMMSS), in 1977; prior to this only material mined in the US was used in processing. At that time, NCA material item tracking and accountability was organized through a "country control number" (CCN). The CCN consisted of specific fields documenting where the uranium was mined, enriched, and irradiated. Ancillary fields were used to identify additional safeguards conditions applicable to the material.

With the maturation of nuclear trade, the US government found the CCN system to be increasingly limited in its ability to capture the evolving nuances in tracking and control of nuclear trade. For example there was no field to identify foreign fabrication of US enriched uranium into fuel. Given the increasing and complex number of NCAs between trading partners, it was decided that the CCN system did not adequately address the growing international trade in fuel cycle services. As a result, the US government in cooperation with the commercial nuclear industry developed a new reporting methodology that satisfied international commitments while reducing domestic reporting complexity. This work led to the current obligation code-based tracking and accounting system, which came into effect in 2003, replacing the CCN system. Under the new system, obligated nuclear material is tracked and accounted for using obligation codes or flags assigned by the US government. Currently, 44 obligation codes exist to account for various nuclear materials subject to one or more NCAs.

As the US official state system of accounting for nuclear materials, NMMSS compiles inventory, transaction, and obligation code data associated with the imports, exports, retransfers, quantity, location, and domestic distribution of obligated nuclear material within the US. This information supports US government adherence to its NCAs and provides assurance to foreign trading partners that nuclear materials are only being used for peaceful purposes.

III. SINGLE VS LAYERED OBLIGATIONS

Using the obligation code-based tracking system, the US can identify whether a quantity of nuclear material is obligated to one NCA (single obligation) or simultaneously obligated to two or more NCAs (layered obligation). An example of single obligation would be an Australian uranium mill shipping

[1] EURATOM includes Austria, Belgium, Bulgaria, Croatia, Cyprus, Czech Republic, Denmark, Estonia, Finland, France, Germany, Greece, Hungry, Ireland, Italy, Latvia, Lithuania, Luxemburg, Malta, the Netherlands, Poland, Portugal, Romania, Slovakia, Slovenia, Spain, Sweden, and United Kingdom

natural uranium (as U_3O_8), to a US conversion plant where the uranium will be converted to uranium hexafluoride (UF_6). The UF_6 would then be subject to the US/Australian NCA and obligated to Australia.

Layered obligations provide significant complexities for receiving countries. Referencing the example above, if the Australian uranium that was converted to UF_6 in the US, was then shipped to France for fabrication into fuel elements, the material would acquire an additional obligation to the US. The uranium would therefore be obligated to Australia and the US.

US facilities are required to add an obligation layer to all nuclear material that passes through foreign obligated equipment or technology. As an example of the impact of using obligated equipment, there are 16 US nuclear power reactors using pressure vessel heads fabricated in Japan. Under the NCA, this equipment is obligated to Japan, and consequently all nuclear fuel, upon use in that reactor, will acquire an obligation to Japan.

IV. OBLIGATION SWAPS

A. *Internal Domestic Swaps*

US facilities frequently adjust the composition of their fuel and material holdings due to government requirements on nuclear trade, demands of reactor operation, and the complexities of the fuel cycle. To reduce the reporting complexities and costs associated with holding multiple single or layered obligated material, US facilities may prefer to hold material with as few obligation codes as possible. However, due to the nature of reactor fuel fabrication, it can be exceedingly difficult to acquire fuel with a single obligation. Therefore, facilities make use of "swaps", or exchanges of nuclear material obligations. Swaps of obligated nuclear material must be identical in quantity, chemical form, and obligation pedigree.

In most cases, obligation swaps are conducted through book inventory adjustments (internal to the facility and reported to NMMSS), avoiding the need to physically transfer nuclear material from one location to another. This results in lower risks of accidents or diversion associated with physical movement of large amounts of nuclear material. Furthermore, swaps are an efficient way for market participants to meet the demand for their services without the cost of transporting either feedstock materials or products. [6]

B. *International Swaps*

An international obligation swap (also called an "international exchange of safeguards obligations" or "international flag swap") is an exchange of obligations between two equivalent quantities of material, located in different countries or jurisdictions. Unlike domestic swaps, international swaps require prior consent from trading partner safeguards authorities. Consent is generally given on a case-by-case basis, and can be time-consuming to complete. For this reason, international obligation swaps are rarely executed.

Through intergovernmental nuclear cooperation agreements, a number of supplier nations have imposed additional requirements beyond those included in the safeguards agreements. Referred to as bilateral agreements, they are used to incorporate specific requirements surrounding exchanges and swaps, and can vary greatly between countries.

Proposed obligation swaps are evaluated using the principle of equivalence embodied in all bilateral and safeguards agreements. A swap proposal must match all obligations attached to the material. Separate lots of obligated material cannot be combined to create a layered obligation and would not be equivalent to a single lot containing multiple obligations. Likewise, layered obligations cannot be "deconstructed" and restructured in order to meet the equivalency requirement.

V. CONCLUSIONS

The complex nature of managing multiple tracking and reporting obligations can lead to less efficient and more costly operations by industries engaged in the nuclear fuel cycle. Swaps provide a safe and economically efficient method of optimizing a country's nuclear fuel cycle procurement strategy, while not diminishing the effectiveness of non-proliferation controls. Through the equivalence principle, an equivalent amount of material remains obligated to supplier trading partner(s) after a swap has taken place, with obligations transferred between quantities of material, not reduced.

The creation of peaceful use obligations on nuclear materials required the investment of significant resources to develop, negotiate, and implement the system of obligation accounting in place today. The agreements and policies developed to implement peaceful use obligations management are purposefully complex and layered to discourage proliferation and detect potential diversion of material to unauthorized uses. However, it can be difficult for nuclear facilities to understand the nuances in NCA language, particularly when it comes to those isotopes presenting the highest risk of proliferation such as highly-enriched uranium or separated plutonium. US nuclear facilities frequently look to DOE and the Nuclear Regulatory Commission (NRC) for guidance with regard to managing as well as complying with foreign obligation restrictions. A potential solution is the development of detailed implementation guidance for peaceful use obligation management at US nuclear facilities. This guidance would support consistency in management of obligations while recognizing the need for efficient, economic facility operations.

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Occupational Exposure Profile and its Implication in the Current Individual Monitoring Program of Nuclear Medicine Workers in the Philippines

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I. Introduction

Nuclear medicine (NM) involves the use of radioactive compounds called radiopharmaceuticals to provide diagnostic information, evaluation, and therapeutic techniques to patients. NM procedures are administered from handling and production of radiopharmaceuticals to injection of radiopharmaceuticals for a variety of diagnostic and therapeutic applications [1]. Over the years, it has become more accessible resulting to an increase in demand. As of 2008, more than 30 million NM imaging procedures were performed worldwide [1]. In USA alone, more than 19 million NM procedures were performed last 2005 [3]. In the Philippines, there were only 20 facilities offering NM procedures as of 2013 which increased to 58 NM centers for 2016 [4].

Nuclear medicine workers handle unsealed radioactive sources. The growing demand for NM procedures and correspondingly the workload of NM workers increase their risk of occupational exposure. It is therefore necessary for them to follow radiation protection measures such as the principle of optimization by keeping the doses as low as reasonably achievable (ALARA) [2]. Due to the associated health risks of exposure to radiation, international standards and the Philippine safety regulations require monitoring and control of occupational exposure. This is to help ensure that the prescribed dose limits are not exceeded [5] [6] [7].

NM workers handle both gamma and beta emitters and are thus at risk of external exposure to the skin, extremities and the whole body. It is therefore important to determine the range of doses they have received as compared to dose limits. This study aims to evaluate the external occupational radiation exposure of NM workers in the Philippines over the years. The results of this study can serve as baseline data for occupational exposure of NM workers in the Philippines. It can also be a means of ensuring whether safety regulations and/or proper radiation protection measures are in place to control occupational exposure.

II. Methodoloy

A. Personnel monitoring devices

The Philippine Nuclear Research Institute (PNRI), through the Radiation Protection Services Section (RPSS), offers individual monitoring services (IMS) to licensed radiation facilities to monitor the workers' exposure to ionizing radiation. The IMS use passive personal dosimeters to measure the amount of accumulated radiation dose received by the workers.

Two kinds of passive dosimeter systems were utilized to measure external occupational exposure, in terms of personal dose equivalents Hp(10) (for whole body effective dose) and Hp (0.07) (equivalent dose to the skin & extremities). The dosimeter systems are provided by the PNRI-RPSS.

For whole body monitoring, thermoluminescent dosimeters (TLD) and optically stimulated luminescence (OSL) dosimeters are used to measure Hp(10) and Hp(0.07). On the other hand, extremity dosimeters utilize ring dosimeters (RD) equipped with TLD to determine Hp(0.07) received by the hands. OSL dosimeters consist of Al_2O_3:C detector elements while TLDs consist of Li:F, Mg, Ti thermoluminescent material. OSL dosimeters are processed using the Landauer InLight Auto 200 Reader while TLDs are processed using the Thermo Scientific Harshaw Model 6600 LITE TLD Reader.

B. Monitored personnel

Personnel monitoring data of NM workers in 2013-2018 were collected from the PNRI-TLD and PNRI-OSL IMS. The assessment of the personal dose equivalents Hp(10) and Hp(0.07) is performed per monitoring period where every monitoring period is equal to 2 months. The annual effective doses were then considered based on the results of monitoring expressed in terms of effective dose Hp(10) and equivalent dose Hp(0.07).

C. Annual doses

The annual occupational dose data of NM workers were presented according to the prescribed online platform by the UNSCEAR as seen in Fig. 1. The online platform is for the Global Survey of Medical Radiation Usage and Exposure of the

Thank you DOST-GIA OneLab Project for the support in the development of the Philippine Dose Registry.

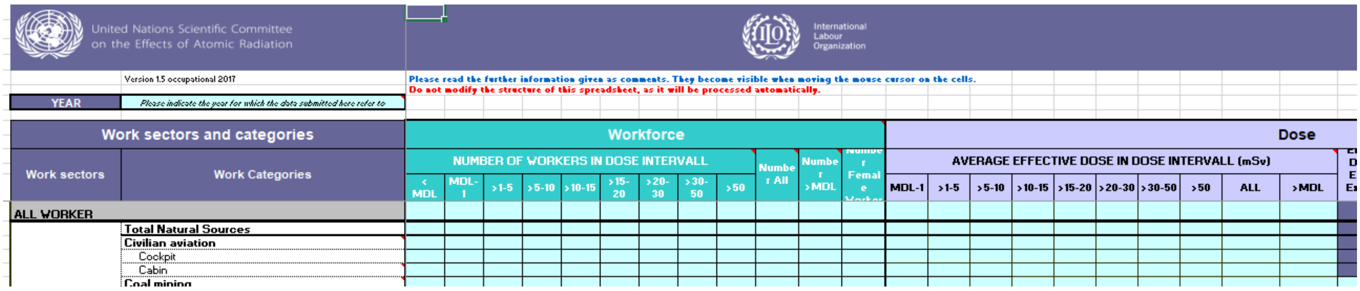

FIGURE 1: Dose assessment online platform for official submissions to UNSCEAR's Global Survey of Radiation Exposure

Global Survey of Occupational Exposure used to provide data on the use of radiation. Annual Hp(10) and Hp(0.07) were arranged according to dose intervals i.e. below minimum detectable limit of the dosimeter (<MDL), MDL-1 mSv, >1-5 mSv, >5-10 mSv, >10-15 mSv, >15-20 mSv, >20-30 mSv, >30-50 and >50 mSv) where MDL is 0.1 mSv.

III. RESULTS

Fig. 2 shows the number of NM workers monitored by PNRI-RPSS's IMS using whole body dosimeters (WBD) OSL and TLD. The workers were grouped according to Hp(10) dose intervals based from the UNSCEAR platform. NM workers include hot lab staff, physicians, NM technologists, medical physicists, and nurses. The population of NM workers and number of facilities had an increasing trend over the years. A total of 60 for 2013, 115 for 2014, 367 for 2015, 334 for 2016, 457 for 2017, and 626 for 2018 NM workers were monitored. The total number of NM workers from 2013 (60 NM workers) increased to as much as ten times (626 NM workers) in 2018. For 2015-2018, there were at least 2 personnel who received annual doses greater than 5 mSv yearly. Per year, at least 49% of recorded NM personnel received annual Hp(10) doses below MDL, while at least 89% of all recorded NM personnel received annual Hp(10) doses below 1 mSv. Table I shows the WBD average annual Hp(10) and Hp(0.07) of all dose intervals versus of those greater than MDL doses. The average annual Hp(10) values ranged from 0-0.59 mSv which is similar to the results of various studies about NM occupational exposure [3] [8] [9]. For the years 2015, 2017, and 2018, there was at least 1 incident of annual Hp(10) dose exceeding the regulatory dose limits.

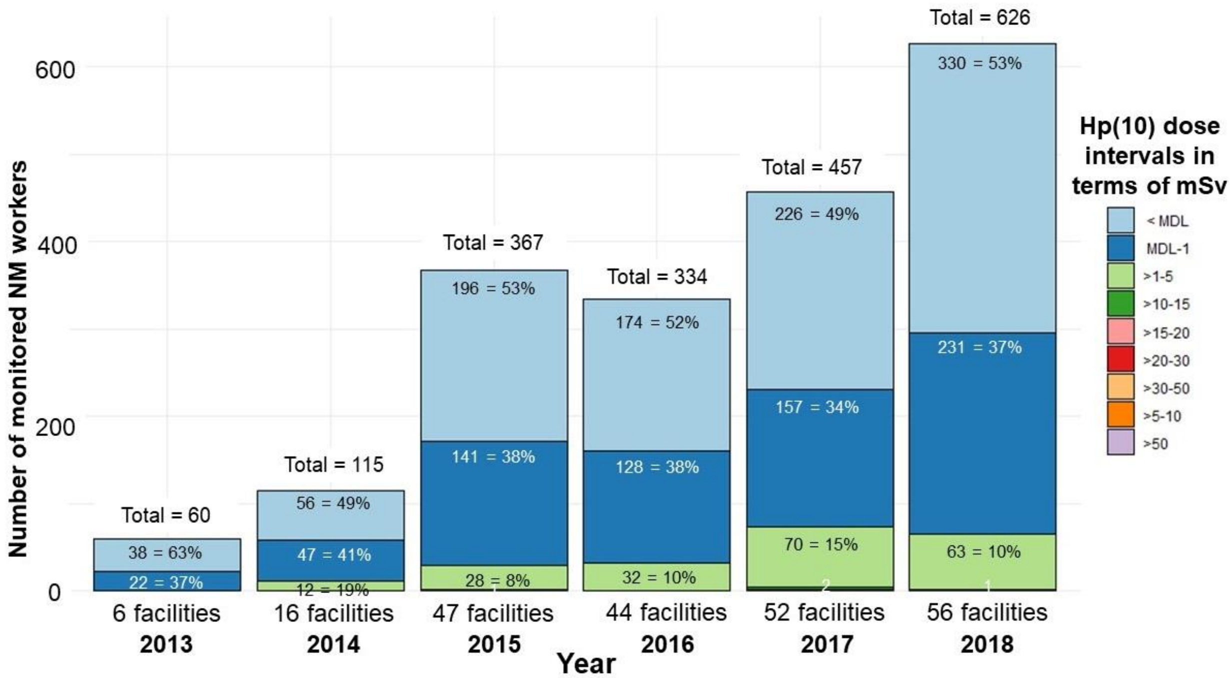

FIGURE 2: Annual population of NM workers in the Philippines monitored by PNRI-RPSS whole body dosimeters grouped according to Hp(10) UNSCEAR annual dose intervals along with annual number of NM facilities

TABLE 1: Average annual occupational doses of Nuclear Medicine workers monitored by whole body dosimeters (WBD) in 2013-2018

Year	Average Hp(10) in mSv		Average Hp(0.07) in mSv	
	All dose intervals	*>MDL*	*All dose intervals*	*>MDL*
2013	0.16	0.33	0.04	0.58
2014	0.37	0.69	0.21	0.94
2015	0.40	0.84	0.19	0.59
2016	0.33	0.67	0.26	0.75
2017	0.54	1.04	0.54	0.91
2018	0.59	1.23	0.64	1.22

All recorded average annual doses were below the regulatory dose limits of 20 mSv/yr averaged over 5 years for Hp(10) and 500 mSv/yr for Hp(0.07) as seen in Table 1 [2]. However, there are still a few recorded incidence of occupational exposure exceeding the limits, with 2018 having the highest recorded annual dose for Hp(10) of 149 mSv and Hp(0.07) of 141.5 mSv both gathered from WBD data. While for the RD, the highest recorded annual Hp(0.07) was 108.8 mSv for the same year.

Table 1 also shows that the average annual Hp(10) and Hp(0.07) of workers monitored by WBD who received greater than MDL is twice as much greater than the average annual dose for all intervals (including those with less than MDL annual doses). This difference is due to the highest recorded doses per year. Removing all the doses below MDL vastly increased the average annual dose.

These results are similar to the study of Alnaaimi et al. [10], hot lab staff of NM facilities can receive occupational exposure up to 140 mSv which is mainly due to exposure during production, dispensing, maintaining, and testing of radioactive materials used in NM procedures. Nurses also tend to receive higher annual Hp(0.07) doses since they always interact with patients [8] [9].

Table 2 shows the comparison between Hp(0.07) garnered from WBD and RD. The average annual Hp(0.07) from RD is greater by as much as ten times compared to the Hp(0.07) from WBD. This shows that the exposure to the extremities are much higher since NM workers deal with direct handling of radioactive sources such as radiopharmaceuticals. The large difference between the Hp(0.07) from WBD and RD presents the higher risk of Hp(0.07) exposure to the extremities thus the necessity for extremity Hp(0.07) monitoring.

Despite this however, extremity monitoring is not widely implemented in the Philippines. The population of NM workers who wear extremity dosimeters like RD are much lower than those who wear WBD (as seen in Fig. 3). As the population of NM workers who use WBD increases yearly, the number of RD users does not. Thus, exposure to the extremities of all the workers is not fully assessed.

Therefore, there is a need to strengthen the individual monitoring program (IMP) in the Philippines. It is recommended that licensees include extremity monitoring of its workers for better assessment of their Hp(0.07) which is also recommended as per international safety standards [11]. The IMS should also enhance its services to ensure that it can provide the demand for extremity dosimeters. The results of this study also show that regulatory bodies may need to revisit the currents requirements of the IMP and determine whether current program are sufficient to assess occupational exposure of NM workers.

IV. CONCLUSION

Since NM workers handle unsealed radioactive sources, it is necessary to track and evaluate their occupational doses to ensure that their exposure is monitored and controlled. This study evaluated the annual occupational doses of NM workers in

TABLE 2: Average annual Hp(0.07) of Nuclear Medicine workers monitored by whole body dosimeters and extremity ring dosimeters in 2013-2018

Year	Average Hp(0.07) in mSv using whole body dosimeters		Average Hp(0.07) in mSv using extremity ring dosimeters	
	All dose intervals	*>MDL*	*All dose intervals*	*>MDL*
2013	0.04	0.58	3.41	3.41
2014	0.21	0.94	9.15	9.36
2015	0.19	0.59	1.96	2.38
2016	0.26	0.75	6.92	7.08
2017	0.54	0.91	5.78	5.78
2018	0.64	1.22	9.04	9.04

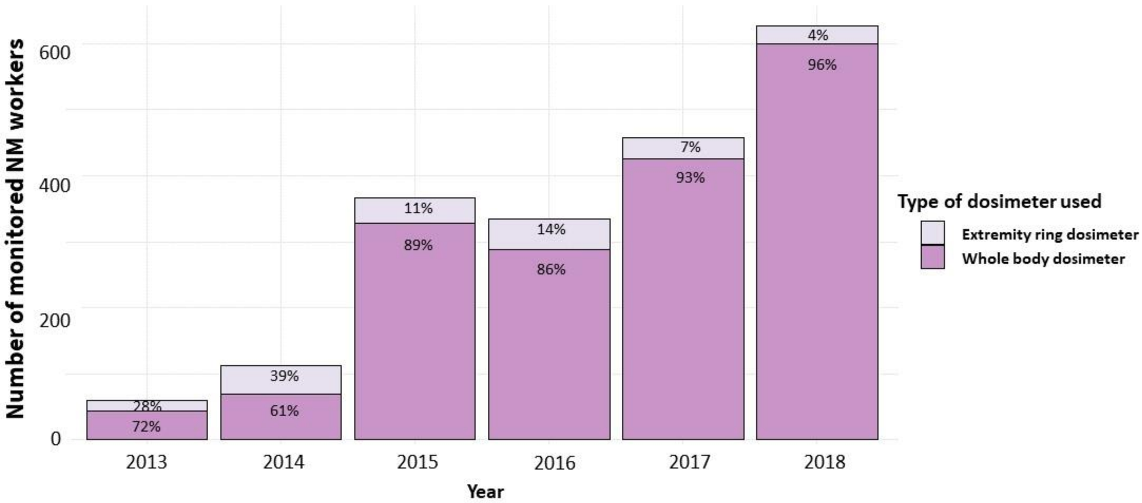

FIGURE 3: Annual population of NM workers monitored by PNRI-RPSS grouped according to the types of dosimeter used to measure the Hp(0.07)

the Philippines for 2013-2018 using WBD and RD. Values for average annual doses were within safe limits. Although most of the recorded annual Hp(10) and Hp(0.07) doses did not exceed regulatory limits, there were a few recorded incidences of doses exceeding limits thus radiation protection measures such as proper shielding, lessening time spent during patient injection among other procedures, and keeping the doses ALARA should be continuously enforced. Hp(0.07) received by NM workers differ depending on the procedure and the type of dosimeter used. The large difference between Hp(0.07) from WBD to RD shows that other staff have higher risks of overexposure in their extremities such as their hands and fingers thus WBD may not be sufficient and the use of extremity dosimeters are highly encouraged and recommended. The use of extremity dosimeters to further assess risks from external exposure may strengthen the IMP in the Philippines. Thus, regulatory bodies should check the requirements of the IMP as they may further improve regulatory and safety programs for occupational exposure of NM workers in the Philippines.

ACKNOWLEDGMENT

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Integration of Research Reactor Computer Security Management with Whole-of-Facility Asset Management Practice

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I. Introduction

The issue of computer security in industrial and nuclear facilities is not new, and has been discussed and reviewed at length for many years. The need to protect Operational Technology (OT) at nuclear facilities as part of a nuclear security regime should be well understood across the industry by the present time. Indeed, significant International Atomic Energy Agency (IAEA) guidance, international and national regulation, frameworks and standards exist across the world. [1]

Computer security operations in mature, corporate computing or Information Technology (IT) environments employ trained specialist security staff following sophisticated defensive processes (*proactive, active* and *regenerative*) and using advanced tools such as Security Information and Event Management (SIEM) or, Security Orchestration, Automation and Response (SOAR) platforms.

The rapid pace of change of OT systems, together with the rapid pace of change in the people, processes and technological aspects of contemporary cyber security operations greatly exceeds that of traditional engineering disciplines. These differences may lead to a divide between OT asset management and cyber security activities and traditional engineering and maintenance activities in terms of technological, administrative, leadership, management etc. It is important that these OT specific cyber security and asset management activities are integrated in a holistic manner with the facilities' overall asset management practice to ensure OT assets are well-maintained and secure in the most efficient and sustainable way.

This paper describes the integrated OT asset management and cyber security program established at the OPAL multi-purpose reactor at ANSTO that has demonstrated significant benefits to cyber security risk reduction, improved reliability and performance, and asset management cost reduction and efficiency.

II. Research Reactor Computer Security Risk Management

Many standards and guides exist in the field of computer security risk management, including standards and guides specific to Industrial Control Systems (ICS) and OT [2,3]. Indeed, IAEA guidance on computer security at nuclear facilities has been available as early as 2011 and 2016 including: Computer Security Risk Management [4]; Conducting Computer Security Assessments; and Computer Security Incident Response Planning. IAEA guidance on advanced computer security risk assessment and management is expected to be published imminently. [5].

Generally, computer security risk assessment requires the following activities to be completed:

1) Identify digital assets and facility functions;
2) Characterise the design basis threat to the facility;
3) Develop attack tree(s) against the identified assets proportinate to the design basis threat;
4) Define security levels and associated security controls comensurate to the developed attack tree;
5) Assign digital assets to security zones within these security levels, applying the respective controls.

After this risk assessment process has been performed, any digital assets currently in operation may then require additional installation or configuration of applicable security controls together with ongoing operational activities as defined by the security risk assessment. Any new systems procured at the facility must be assessed to identify the applicable security level, zone and controls that are required to be implemented on the system, and the configuration and resulting effectiveness of those controls must be included as fundamental requirements of the design and procurement of the system (cyber-informed engineering) [6]. For example, the need to include additional capacity or redundancy in an OT system beyond the purely functional or operational capacity requirements to allow for security maintenance activities to be performed such as routine patching.

III. Asset Management

At a high level, the ISO 55000 series asset management standards [7] have many similar objectives and activities as cyber security risk management standards and guides. Both require the identification and characterization of assets, assessment of the assets' role in value creation to the organisation (e.g. safety, reliability), the expected operational requirements of the asset, financial characteristics of the asset (e.g. opex costs, depreciation, planned capex costs), and appropriate ongoing maintenance and long-term asset management tasks required to maintain the asset.

When characterizing the ongoing maintenance activities required to maintain and asset's reliability, Reliability Centered Maintenance (RCM) principles are used. RCM principles require a characterisation of a system and its components, and a detailed failure modes and effects analysis (FMEA) to be completed, before then describing five risk management strategies to be applied to the asset and its components:

- Predictive maintenance tasks;
- Preventive maintenance tasks;
- Detective maintenance tasks;
- Run-to-Failure, and
- Design changes.

The RCM FMEA process for physical plant systems is similar to an attack tree analysis process for OT assets. Where an FMEA for a mechanical asset will consider physical and environment "threats" to an asset e.g. temperature, pressure, flow, an attack tree analysis to an OT asset will consider threat actor actions e.g. vulnerability exploitation, replay attack, code injection.

Additionally, cyber security controls applied to OT systems can also be identified with each of the five RCM risk management strategies:

- Predictive maintenance tasks: threat detection systems and processes, vulnerability management systems and processes;
- Preventive maintenance tasks: operating system and application software patching, whitelisting, hardening;
- Detective maintenance tasks: OT/IT condition monitoring, SIEM, threat hunting and incident response activities;
- Run-to-Failure: utilising a honeypot strategy; and,
- Design changes: proactive cyber-informed re-engineering of OT systems.

IV. INTEGRATION

By including the cyber security risk management processes for a facilities' OT systems in the whole-of-facility asset management practice, several efficiencies are realised:

1. The use of cyber-informed engineering processes to include cyber security requirements in early system design and development reduces the amount and complexity of retrofitting cyber security controls on to a system in later stages of project delivery or post-commissioning

2. Considering ongoing cyber security control activities as *maintenance* activities (rather than as separate "special cyber security or IT activities") and including these activities within standard maintenance plans and processes, using the same terminology, simplifies communication between OT and other team members.

3. Tracking compliance to these maintenance plans using the same methodology as all other maintenance plans simplifies reporting and understanding of decision makers and other stakeholders.

The realisation of these efficiencies in-turn allows for improvements to overall cyber security posture and mitigation of cyber security risk to the facility and its operations.

ACKNOWLEDGMENT

Many thanks to Anthony, Tina, Julian and the rest of the very professional staff in Reactor Operations for their support and friendship over the many years taken to get our OT asset management and cyber security practice where it is today.

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FLEX Strategies analysis under a LBLOCA scenario with the MELCOR code

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I. Introduction

Nowadays, nuclear safety is a discipline that uses Fukushima accident lessons to improve. The loss of core cooling capability in that accident lead to core melting, hydrogen combustion and vessel failure, creating an incident that changed the paradigm of nuclear energy worldwide [1].

After the accident, the international nuclear community made a strong effort in applying the lessons learnt from Fukushima to all others reactors. Thus, in order to reduce the risk of loss of core cooling capability, the Diverse And Flexible Coping Strategies (FLEX) emerged by the hand of the Nuclear Energy Institute (NEI), [2]. Some of the FLEX strategies are aimed to prevent the Vessel Failure (VF) during a severe accident (SA). If VF is avoided, the consequences of the accident can be strongly diminished. For this reason, FLEX strategies cope with this challenge named In Vessel Retention of molten corium (IVR) by trying to create a cooling path, internal, and/or external [3].

One of the main pillars of these strategies is the use of portable equipment that provides means of obtaining power and water to maintain and/or restore key safety functions for all reactors at a site, with a reasonable staging and protection of portable equipment during a Severe Accident (SA) applicable to the site. These strategies are currently implemented in most of the reactors around the world and in the US, they became mandatory for modification to licenses and construction permits after 2012 [4]. The use of portable equipment permits the recovery and/or support of safety systems. Figure 1 provides an example of one of these strategies: external pumps injecting water into the reactor pressure vessel. Based on this portable equipment performance, the accident progression can be modified, but more research in this field is needed.

In a previous study of Gómez et al. [5], different accident timings and FLEX actions are analyzed in a German Konvoi reactor with the ASTEC code during a MBLOCA. Similarly, Xiao et al., [6] studied a SBO scenario in a PWR with the code MAAP.

Following this trend, the present study will analyze a Large Break LOCA (LBLOCA) with Emergency Core Cooling System (ECCS) failure during the recirculation phase, with not only different timing for the recirculation failure, but also different timing for FLEX implementation. This study is performed with the MELCOR 2.2 code in a PWR-W with the FLEX strategies focused on injecting water into the core after its degradation.

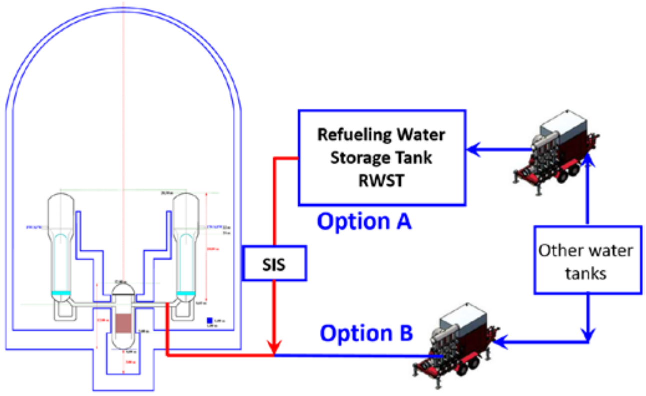

Figure 1. Example of different options for FLEX portable equipment injection.

The present paper is divided in three additional sections. The next section is dedicated to the MELCOR code and the models used for the analysis. Following, a depiction of the scenarios without any FLEX strategy is presented and then, the analysis of the FLEX strategies applied to the current case. Finally, some conclusions are drawn.

II. Computational Model

A. The MELCOR code

The MELCOR code is a fully integrated, engineering level computer code, capable of modeling accident phenomena in Light Water Reactors (LWRs). It was developed by Sandia National Laboratories (SNL) for the U.S. Nuclear Regulatory Commission for SA analysis. The MELCOR capabilities include the simulation of thermal-hydraulic behavior of the reactor coolant system (RCS) and the containment, core damage progress including in-vessel melt progression and relocation, molten core concrete interaction, behavior of fission products, hydrogen generation and combustion, among others, [7]. MELCOR code contains models for core quenching after degradation. Moreover, core quenching after degradation, remains one of the most computationally demanding and complex phenomena of a SA, being the focus of many recent studies, i.e. QUENCH project [8]; therefore, the results obtained will have to be interpreted accordingly.

B. PWR-W MELCOR model

The MELCOR model used for the present study is a PWR-W with 3 loops. The model follows the best practice guidelines recommended in [9,10] in terms of core nodalization and core parameters. This model is an evolution of a previous PWR-W model developed in the UPM, [11]. The model main characteristics can be found in TABLE 1, and the RCS nodalization is shown in Figure 2. The model includes explicit representation of the entire RCS including each of the reactor loops, the pressurizer relief tank, the Steam Generators (SG), steam lines until the isolation valves, and associated safety and power-operated relief valves. The containment is divided into 51 control volumes where the 30 spray components are distributed.

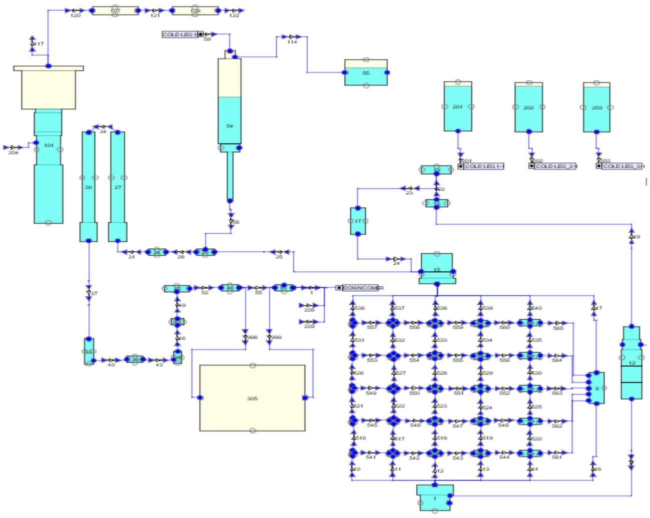

Figure 2. Nodalization of the coolant system of the PWR MELCOR model

Water injection provided by the FLEX equipment is modelled as a constant flow rate into the cold leg at a constant temperature. The recirculation phase of the safety and/or portable systems is modelled by extracting water from the containment sump and at the same rate, injecting it in the cold leg at a constant temperature to recreate the heat exchanger present in the PWR-W model. The FLEX injection, is located at the injection of the primary SIS.

TABLE 1 MAIN CHARACTERISTICS OF THE PWR-W MELCOR MODEL

Control Volumes	Heat Structures	Core Axial Levels	Core Radial Rings	Cavities	Decay Heat
135	275	13	6	1	ORIGEN code
RN Release Model	RN Classes	User defined NCG	Flow Paths	Hydrogen Burn	Containment Sprays
Corsor–M with S-V ratio	17	9	284	Disabled	30

III. LBLOCA RECIRCULATION FAILURE WITHOUT FLEX STRATEGIES

To study the FLEX strategies, the selected accident is a LBLOCA with failure of the ECCS during the recirculation phase. The safety injection starts normally right after the LOCA occurs. The LOCA is modelled as a guillotine break in the cold leg, reaching a discharge peak of 30000 kg/s. When the Refueling Water Storage Tank (RWST) reach a depletion setpoint value (with enough margin to avoid the pumps cavitation), the LPSI system change from injection mode to recirculation mode. This occurs 2500 seconds after the break, as shown in Figure 3. Following the recirculation failure, the liquid level of the core starts to decrease, while the temperature in the fuel starts to rise as soon as it gets uncovered. When the fuel surpasses around 2500 K, it starts to melt, candling downwards and re-freezing again. When the amount of molten core material candling downwards is enough, it can reach the vessel lower plenum, creating a molten pool. Finally, the reactor pressure vessel fails by exceeding the mechanical stress limits, or when the penetrations' failure temperature is reached, spreading the corium into the reactor cavity. At this point, the simulation is stopped, and no further development of the accident is studied.

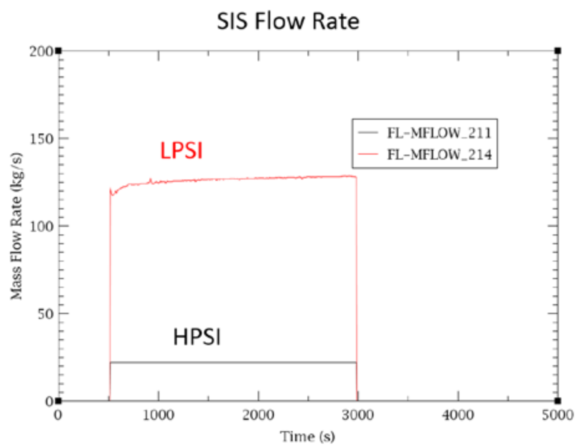

Figure 3. SIS mass injection flow rate.

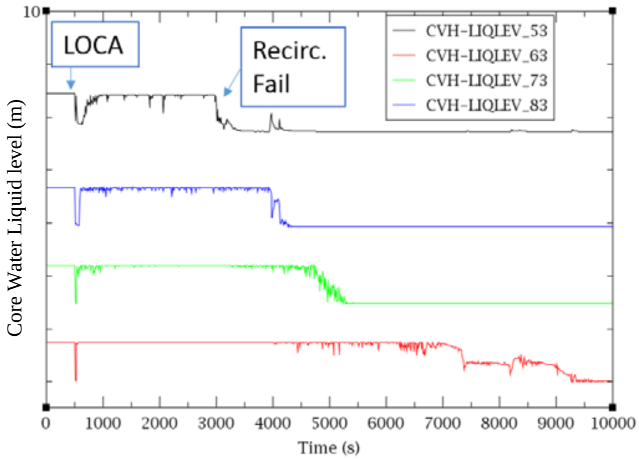

Figure 4. Liquid Level in some Control Volumes of the Core.

In the present simulations, the time of the recirculation failure is varied between 0 to 26000 s after the recirculation starts 2500 seconds after the break. From the results obtained, (shown in Figure 5) it is seen that if recirculation fails on demand, the cladding melting appears at 3600 s, the relocation into the lower plenum at 5200 s and the vessel fails 11600 s after the recirculation mode failure These results are in agreement with previous studies on ECCS failure in the recirculation phase of a LOCA, [5]. Additionally, it is seen that the time between the initial core damage and VF timing is slightly increased as the recirculation failure is delayed in time. Additionally, the relocation of corium into the lower plenum gets significantly delayed during the accident as the recirculation failure delays. It is worth remarking the different time range of the axis of Figure 5, as it can be seen that the later the recirculation failure, the later the core degradation process begins; this is due to the decrease of the decay heat lowering with time.

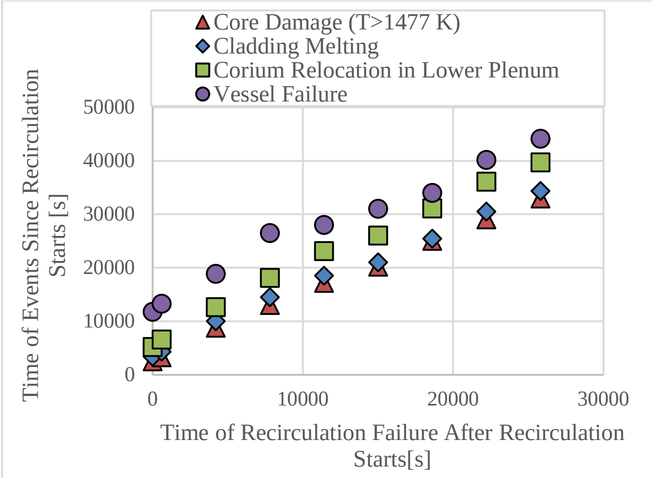

Figure 5. Accident progression, relative to the recirculation failure time.

IV. LBLOCA Recirculation Failure with FLEX Strategies

The simulation is repeated with a modified scenario, i.e. a portable water injection pump is supposed to be available after some time following the recirculation failure.

When water gets in contact with a degraded core, a rapid evaporation occurs, forming steam, which can oxidize the cladding and produce a great amount of hydrogen. For this reason, core reflooding has to be performed with a sufficient water flow rate to quickly quench the full core, and avoid more oxidation. In this sense, if the core degradation is not too large, it would be possible to stop it.

The different combinations of Recirculation Failure times and FLEX injection starts are shown in Table 2, with the legend explained in Table 3. In Table 2, the color indicates the damage state that the FLEX injection encounters at its start, and the wording indicates the final state. The mass flow rate provided by the portable pump is assumed to be 20 kg/s.

Table 2. LBLOCA sequence. Core Damage State for different recirculation failures and FLEX injection timings.

G = 20 kg/s		Recirculation failure since recriculation begins								
		0 min	10 min	70 min	130 min	190 min	250 min	310 min	370 min	430 min
FLEX Inyection time since failure	30 min	SS	SS	SS	SS	SS	SS	SS	SS	SS
	60 min	FM	FM	SS	SS	SS	SS	SS	SS	SS
	90 min	CR	CR	FM	FM	FM	CD	SS	SS	SS
	120 min	CR	CR	FM	FM	FM	FM	FM	CD	CD
	150 min	CR	CR	CR	FM	FM	FM	FM	FM	FM
	180 min	CR	CR	CR	CR	CR	CR	FM	FM	FM
	210 min	VF	CR	CR	CR	CR	CR	CR	CR	CR
	240 min	VF	VF	VF	CR	CR	CR	CR	CR	CR
	>270 min	VF	VF	VF	VF	VF	VF	VF	VF	VF

TABLE 3. LEGEND OF COLORS AND NAMES FOR CORE DAMAGE STATES OF TABLE 2.

Safe State (SS)	
PCT > 1477 K (CD)	
Fuel Melting (FM)	
Corium Relocation (CR)	
Vessel Failure (VF)	

In Table 3, FM is defined as the moment of fluidization, based on a Time-Temperature relationship, as [9,10]. The CR is defined as the moment where more than 5 kg of UO2 are found in the lower level of the core.

In Table 2, it is seen how different core damage states are affected by the timing. It is seen that under some circumstances, the FLEX injection is not capable of stopping the accident progression. It is also confirmed, as in previous studies, that if more than 25 tons of corium are relocated in the lower plenum, the vessel failure is unavoidable, see [5]. Additionally, it is seen that after 270 min with no FELX injection, it is not possible to avoid VF. As a final observation, it is seen that it is better to inject water into the RCS as soon as possible, instead of delaying the failure of the recirculation.

V. CONCLUSIONS

The FLEX strategies are being implemented in the Emergency Operating Procedures and SA management guidelines of most of the reactors around the world. Within these strategies, the use of portable equipment is proposed as an adequate accident management measure to prevent or mitigate the consequences of a SA following a loss of core cooling capability scenario.

In the present research, a study of the different timings of water injection into the RCS, together with the failure of the safety injection systems during the recirculation phase of a LBLOCA scenario in a PWR-W are analyzed using the MELCOR 2.2 code.

The study shows that core damage may be precluded and/or stopped when the appropriate combination of water injection starting time and recirculation failure time are considered. Additionally, it was found out that to inject water into the RCS as soon as possible, is a much safer option than to delay the onset of the recirculation phase failure.

The present research acts as another step in the necessary study of both FLEX strategies and core quenching of a degraded core. This work is intended to be extended to more equipment capabilities and scenarios, to help in the shaping of the final form of FLEX strategies.

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The role of nuclear forensics in supporting the peaceful application of nuclear science and technology: an Australian perspective

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I. Nuclear in Australia

The Australian nuclear industry is quite unique with a great concentration on the front end of the nuclear fuel cycle. Australia is home to 30% of the world's uranium resources, with more than double the deposits of the next most abundant nation [1]. With over 6500 tonnes of U_3O_8 mined in 2018, Australia is the third largest producer, with the entirety of the product being exported as uranium ore concentrate for enrichment and fuel production [2].

In contrast, Australia is home to only one nuclear reactor, the open pool Australian light-water reactor known as OPAL, commissioned at ANSTO in 2007. OPAL is a multipurpose reactor which in 2018 produced 46% of the world's neutron transmutation-doped silicon for semiconductors and a large amount of the medical diagnostic isotope technetium-99m amongst many other medical isotopes [3]. The generation of nuclear power and the processes of enrichment and fuel reprocessing are banned under federal Australian law.

An indigenous nuclear industry and our place in a region and world abundant in nuclear technologies requires the support of a nuclear security architecture. In reality, no nuclear-related activity would be able to operate without the regulation, protection and social license that is provided by a diverse range of nuclear security mechanisms. One of these enabling functions is provided by a nuclear forensics capability with its role in ensuring nuclear material is possessed by authorised people for peaceful applications as well as helping to protect our borders from the entry of unauthorized material from other countries. The existence of nuclear and radioactive material out of regulatory control could imply interest in the production of nuclear weapons or improvised radiological dispersion devices, but often events are motivated by curiosity in nuclear materials and a lack of understanding of the hazards.

II. Australia's Nuclear Security Architecture

In Australia, the security of the nuclear industry and our nation from internal and external threats is governed by a framework of organisations with varied yet interlinking functions:

- The application of safeguards, the protection and security of nuclear material, and Australia's international obligations in the areas of non-proliferation and safeguards are the responsibility of the Australian Safeguards and Non-proliferation Office (ASNO).
- Radiation safety regulation and management of licenses for possession and operations of nuclear facilities is governed by the Australian Radiation Protection and Nuclear Safety Agency (ARPANSA).
- Monitoring our borders for illicit transport of nuclear and radioactive material is conducted by the Australian Border Force (ABF).
- Coordination of Australia's non-proliferation initiatives including export control, arms control treaties and prevention of proliferation of weapons of mass destruction is partaken by the Department of Foreign Affairs and Trade (DFAT), supported by Department of Defence.

Another vital operational force in ensuring Australia's nuclear security is the nuclear forensics capability operated at ANSTO, utilised by the Australian Federal Police (AFP).

III. What is Nuclear Forensics?

Nuclear Forensics is defined by the International Atomic Energy Agency (IAEA):

"Nuclear Forensics is the examination of nuclear or other radioactive material, or of evidence that is contaminated with radionuclides, in the context of legal proceedings under international or national law related to nuclear security" [4]

In the well-established traditional forensics disciplines of DNA and fingerprints, experts analyse these forms of evidence to gain information in support of criminal investigations. In the same way, nuclear forensic scientists analyse evidence either composing of or contaminated by nuclear or radioactive material that is found outside of regulatory control to support investigations.

The requirement for a nuclear forensic capability stems from a need to enforce laws and regulations that govern the possession, movement and peaceful applications of nuclear science and technology and to understand threats or weak links in nuclear security. Using speeding as an analogy, would the public be inclined to obey the laws in regards to speed limits if there was no radar technology to prove their non-compliance? As in the role of speed monitoring technology, nuclear forensics plays a role in deterrence, by simply demonstrating the capability to prove criminal behaviour, and also in response to nuclear security events.

IV. Application of Nuclear Forensics in Australia

The nuclear forensic capability of Australia is held by ANSTO in collaboration with the AFP and its experts in analysing traditional forensic evidence. The capability has been evolving over almost two decades, stimulated by increased awareness of the potential use of such material by malicious actors. The initial focus of the capability was nuclear material analysis and border security technologies. The capability then developed with a body of research into the effects of radiation on various evidence types and the decontamination from radioactive material of these exhibits for safer analysis, especially DNA evidence [5], [6], [7], [8].

The aim of a nuclear forensic examination is to answer a range of investigative questions posed by law enforcement or obtain information relating to other nuclear security concerns. These may be very broad, such as “what is it?” or more specific, such as “how much is there?” or “where did it come from?” The ultimate aim for law enforcement is usually to answer the question: “Has a crime been committed?” ANSTO contributes to answering these questions by accommodating the analysis of radioactive material or of other evidence types contaminated with radioactive material that regular forensic laboratories are not equipped to handle.

A nuclear forensic examination can reveal a large amount of different information about a wide variety of sample types. For example, examination of a nuclear fuel pellet could determine: isotopic ratios and therefore enrichment; whether the uranium used to make the fuel has been inside a nuclear reactor; the age of the uranium used to produce the fuel by measuring parent/daughter radionuclide ratios; and how the fuel was produced, based on trace level contaminants either carried from the starting material or introduced during the production process.

In the Australian context, much of the expertise required is around the less processed materials such as ores and uranium ore concentrates (UOCs) due to the scale of uranium resources and mining. For these sample types it may be possible to determine: methods of processing; when the ore was concentrated (age dating); the type of mineral deposit it came from; uranium isotope ratios; and even specific patterns in abundance of the lanthanide rare earth elements. The combination of these signatures can form a fingerprint-unique picture which may be able to link a UOC to a specific mine and period of production with the aid of reference samples [9]. Medical and industrial radioactive materials can also be examined to ascertain the manufacturer, original use and licensed user where it is found outside of regulatory control.

The scope of analytical techniques that can be applied to a nuclear forensics examination is extremely broad and both technical and subject matter expertise from across the nuclear industry is invaluable. Radiation protection specialists enable examinations to occur safely; nuclear fuel cycle experts provide context and meaning to material analysis; and nuclear physicists, engineers and chemists support key examinations such as gamma spectrometry, imaging and mass spectrometry. Some examples of analytical techniques that can be applied to examinations of nuclear and radioactive material, and the information they can provide are presented in TABLE I. The emboldened techniques are operated at ANSTO. In addition to nuclear material analysis, the nuclear forensics facilities at ANSTO, with purpose-built gloveboxes for containment, accommodate the AFP forensic specialists to process DNA, digital and fingerprint evidence.

TABLE I. Examples of analytical techniques that may be applied for nuclear forensics

Characteristic/ Material Signature	Analytical Technique*
Physical characteristics: shape, dimensions, weight, colour, density	**Visual inspection & photography** Physical measurement tools: (**balances, calipers,** micrometers); **pycnometry** Imaging tools: (**x-ray radiography, neutron tomography, CT**)
Microstructure and morphology	**Optical microscopy, SEM, TEM**
Elemental composition	**XRF, SEM-EDS, ICP-AES, ICP-MS, NAA, DNAA, ion beam analysis techniques**, SIMS
Chemical phase identification	**XRD, neutron diffraction, TEM, SEM-EBSD**
Isotopic composition	**Gamma spectrometry, alpha spectrometry, ICP-MS, AMS,** SIMS, TIMS
Age dating (radiochronometry)	**ICP-MS, alpha spectrometry, AMS**, TIMS, **gamma spectrometry**
Anionic composition	**Ion chromatography**

* Acronyms defined in Glossary section below.

As nuclear forensic scientists, it is not possible to be experts in all of the disciplines that can be used to inform an examination. In the Australian model, nuclear forensics staff at ANSTO are primarily trained in forensic science or chemistry and utilise a large suite of analytical instruments across the organisation as well as associated technical and subject matter expertise. The role of the nuclear forensics team involves: liaison with law enforcement to understand investigative questions; analytical planning of the analyses required to answer the questions; and collating and interpreting analytical data to draw conclusions. The other important responsibility is in protecting the integrity of the evidence by maintaining chain of custody and other documentation that supports the expert testimony to be admissible in court, as well as testifying as an expert witness.

V. Strength in Cooperation

In order to uphold an efficient, effective and sustainable nuclear forensic capability, collaboration and cooperation with a variety of organisations, regional partners and scientific communities is imperative. Domestically the nuclear forensic capability at ANSTO works with state and federal law enforcement, Defence, Border Force and other forensic experts to ensure operational readiness of the nation in a nuclear security event. This involves memorandums of understanding, knowledge sharing, radiation protection and radiological crime scene training as well as readiness exercises and analytical collaborations with AFP forensic experts. Australia is regarded as a leader in the South-East Asian region in the field of nuclear forensics and has facilitated training programs to strengthen the capabilities of the region. The Australian nuclear forensics team also works closely with international organisations and scientific communities such as the IAEA and the International Technical Working Group on Nuclear Forensics (ITWG) among others to share expertise and partake in exercises to ensure the capability is on par with best practice in the international community.

VI. Conclusion

Despite the many bounds placed on its nuclear industry Australia is home to a rich and diverse array of nuclear expertise and technologies. Our place in the highly connected and rapidly changing world and our abundance of nuclear resources means an operational nuclear forensic capability is an important function of an effective nuclear security framework. These actions in both deterrence and in proving criminal misconduct will continue to be vital in stewardship of an Australian nuclear future that may look very different to today.

Glossary

AMS	Accelerator mass spectrometry
CT	Computed tomography
DNAA	Delayed neutron activation analysis
ICP-AES	Inductively-coupled plasma- atomic emission spectroscopy
ICP-MS	Inductively-coupled plasma- mass spectrometry
NAA	Neutron activation analysis
PIXE	Particle induced X-ray emission
SEM	Scanning electron microscopy
SEM-EBSD	Scanning electron microscopy- electron backscatter diffraction
SEM-EDS	Energy-dispersive X-ray spectroscopy
SIMS	Secondary Ion Mass Spectrometry
TEM	Transmission electron microscopy
TIMS	Thermal Ionisation Mass Spectrometry
XRD	X-ray diffraction
XRF	X-ray fluorescence spectroscopy

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When Radiation Protection and Occupational Hygiene Meet: A Case Study on Uranium

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I. Introduction

The Australian Nuclear Science and Technology Organisation (ANSTO) is one of Australia's largest research organisations. At ANSTO we undertake a wide range of activities that benefit Australia through the application of nuclear-based science. Work with uranium ore and metal processing is one of the many activities encompassing research and development at ANSTO. Such activities commonly involve uranium compounds in various forms.

Expectedly, controlling exposure to radiation is often identified as one of the highest priorities during these activities and under Australian Work Health and Safety (WHS) legislation workplaces should keep worker exposures to uranium as low as reasonably practicable [1, 2]. However, different workplace exposure limits exist on the basis of evaluating the effects of both the chemical dose and the radiation dose.

It is important to recognise that the methodologies used in the calculation of radiological and chemical exposure limits are different. As a consequence of the different approaches, adherence to one limit may not necessarily mean compliance or protection with the other. The purpose of this article is to raise awareness of the application of different workplace exposure standards for chemical hazards and radiation hazards in the workplace and the importance of workplace exposure monitoring in determining compliance with these standards.

II. Occupational Health Management

ANSTO's occupational health risk management aims to recognise, evaluate and control risks to health across all our program disciplines including Occupational Hygiene, Health Physics and Radiation Protection. The elimination or control of agents that pose a health hazard, including: chemical hazards, physical hazards, biological agents and ionising and non-ionising radiation is a priority. Uranium is a member of a small group of agents that are chemotoxic and radiotoxic, as such both these properties need to be taken into consideration when assessing occupational exposure and compliance with workplace exposure limits.

III. Uranium

Uranium is a naturally occurring silvery-gray, soft metallic element. Uranium compounds have a range of colours including yellow, green, red and black [3]. Uranium may occur in the environment from the leaching of soils, rocks and natural deposits, release in mill tailings, combustion of coal and other fuels, and use of phosphate fertilisers [4]. Natural uranium contains a mixture of U-234, U-235, and U-238. All decay by alpha particle emission, with associated gamma rays [3]. Natural uranium consists of 99.28% U-238 by weight, with the other isotopes being less than 1% [3]; but of only 48.3% U-238 by activity, with U-234 and U-235 providing the other 49.5% and 2.3%, respectively [5].

IV. Health Hazards Associated With Uranium

The effects of exposure to any hazardous substance are determined by the dose i.e. the amount of the substance, the time exposed and the effectiveness of any controls. Reviews by the Agency for Toxic Substances and Disease Registry (ATSDR), World Health Organisation (WHO) and American Conference of Governmental Industrial Hygienists (ACGIH) have been published and describe the kidneys as the most affected organ from uranium exposure [5, 6, 7]. These health effects are often attributed to heavy metal toxicity rather than radiotoxicity [6]. However, it has also been reported that the chemical and radiotoxicity may have a dual mode of action and the health effects associated with both hazards, overlap i.e. additive or synergistic effects [6]. In terms of occupational hygiene, inhalation is the main route of exposure in industry, although other routes (ingestion and skin) should still be controlled [5, 6, 7]. In terms of radiation protection, the routes of exposure include both the external radiation hazard and the internal radiation hazards, the former being direct gamma irradiation from sources external to the body, and the latter being situations where the source may be taken into the body, such as inhalation of long-lived alpha emitters in dust, inhalation of short-lived radon decay products, and ingestion.

The Safe Work Australia (SWA) Hazardous Chemical Information System (HCIS) provides the following classification as to the health effects for uranium and its compounds (Table I). This classification information for chemicals is taken from the European Union's Classification, Labelling and Packaging (CLP) Regulation [8].

TABLE I. HAZARDOUS CHEMICAL DETAILS SOURCED FROM SWA [8].

Uranium (7440-61-1)	
Globally Harmonised System (GHS) Classification	***Hazard Statement***
Acute toxicity: Oral: Category 2	H300 (Fatal if swallowed)
Acute Toxicity: Inhalation: Category 2	H330 (Fatal if inhaled)
Specific target organ toxicity - Repeated exposure, Category 2	H373 (May cause damage to organs through prolonged or repeated exposure)

V. OCCUPATIONAL EXPOSURE LIMITS

Government regulators worldwide have also made recommendations which apply to Uranium. The Occupational Exposure Limits (OELs) for Uranium worldwide are variable, which can add complexity to assessing compliance. In Australia, the Workplace Exposure Standard (WES) for "Uranium (natural), soluble & insoluble compounds (as U)" that must not be exceeded is 0.2 mg/m^3 eight hour Time Weighted Average (TWA) and 0.6 mg/m^3 Short Term Exposure Limit (STEL) [8] to meet duties under the Work Health and Safety (WHS) Act and the WHS Regulations. These WESs are derived from the American Conference of Governmental Industrial Hygienists (ACGIH) version of the Documentation of Threshold Limit Values (TLV) and Biological Exposure Indices, 6th Edition, 1991 [8]. The 2001 ACGIH TLV values for "Uranium (natural), soluble and insoluble compounds, as U" also still stand although on the ACGIH "under study list" [5].

ACGIH guidelines are health based developed from the review of scientific and toxicological information, and provide regulators and industry groups proposed benchmarks for establishing OELs. The TLV for uranium has been established as a safe level to prevent injury to the kidneys and other organs. The various international limit values for uranium established by various work health and safety institutions are presented in Table II [9]. The OEL for soluble uranium compounds is lower in some countries as it is considered that these compounds are more likely to be absorbed by the body and represent a greater exposure risk than insoluble uranium compounds.

The ACGIH also recommends a Biological Exposure Indice (BEI) value for all forms of natural uranium of 200 µg/L in urine obtained at the end of shift. The BEI is only valid after at least 60 days of uranium exposure (current 2019), because the elimination half-life from the kidneys is long [4].

Based on recommendations from the International Commission on Radiological Protection (ICRP), the Australian Radiation Protection and Nuclear Safety Agency (ARPANSA) set the regulatory radiation dose limits as no more than 1 mSv per year effective dose for members of the public, and no more than 20 mSv per year effective dose averaged over five years for occupationally exposed persons, with the further provision that the effective dose must not exceed 50 mSv in any single year [11]. However, these limits apply to the total dose received from all exposure routes, therefore must account for exposure due to direct irradiation, inhalation of long-lived alpha emitters, inhalation of short-lived radon decay products and ingestion. These are all assessed separately as they require different methods of monitoring and analysis. The inhalation of long-lived alpha emitters is the focus in this discussion for the purposes of comparison with the main occupational hygiene exposure route of inhalation.

Calculation of internal radiation doses requires the use of dose conversion factors, which are based on the radionuclides involved and their particle size and solubility. These dose conversion factors are provided in the International Atomic Energy Agency (IAEA)'s International Basic Safety Standards [10]. These conversion factors can then be used to calculate the Annual Limit on Intake (ALI) of a radioisotope, which is the total activity (Bq) which inhaled over a working year will give a person a dose equal to the annual dose limit. The ALI is then used to calculate the Derived Air Concentration (DAC), which is the maximum allowed airborne activity concentration (Bq/m^3) that can be inhaled over a 2,000 hour working year. DACs can be used as comparison values against monitoring results but they are not published as permissible exposure limits in the same way that TWAs are and should not be used as direct compliance indicators, as they do not take into account what relative contribution to the total dose is provided by the inhalation route compared to the other exposure routes, nor the regulatory requirement for optimization of doses to as low as reasonably achievable (ALARA) below the regulatory limits. These assessments must be made on a site-specific basis in order to select a suitable airborne activity concentration limit.

For example, the DAC for uranium ore dust is 2.4 αdps/m^3 using dose conversion factors taken from ARPANSA RPS 9 [12], but this value presumes the total radiation dose will be received from the inhalation route, when in reality it may only account for a fraction of the total dose. A suitable fraction of this DAC should be used as an investigation trigger level instead.

TABLE II. INTERNATIONAL WORKPLACE EXPOSURE LIMITS FOR URANIUM (NATURAL) [9]

International Workplace Exposure Limits (mg/m^3)				
Country	***Uranium compounds, natural, insoluble***		***Uranium compounds, natural, soluble***	
	TWA	STEL	TWA	STEL
Australia	0.2	0.6	0.2	0.6
Austria			0.25, inhalable aerosol	1, inhalable aerosol
Belgium			0.2	0.6(1)
Canada-Ontario	0.2	0.6	0.2	0.6
Canada-Quebec	0.2	0.6	0.05	
Denmark			0.2	0.4
Ireland			0.2(1)	0.6(1)
Germany (DFG)	0.2	0.4		
Latvia	0.075		0.015	
New Zealand	0.2		0.2	
Singapore	0.2	0.6	0.2	
South Korea			0.2	0.6
Spain			0.2	0.6
Switzerland			0.2 inhalable aerosol	
USA – NIOSH	0.2	0.6	0.05	
USA – OSHA	0.25		0.05	
United Kingdom			[0.2]	[0.6]

(1) 15 minutes average value

(2) United Kingdom: The UK Advisory Committee on Toxic Substances has expressed concern that, for the OELs shown in parentheses, health may not be adequately protected because of doubts that the limit was not soundly-based. These OELs were included in the published UK 2002 list and its 2003 supplement, but were omitted from editions published from 2005 onwards.

VI. EXPOSURE ASSESSMENT METHODS

Assessment of uranium in the workplace should consider both chemical toxicity and radioactivity to determine compliance with OELs. Air monitoring is a specialist activity that should be undertaken by a qualified Occupational Hygienist, Health Physicist or Radiation Protection Advisor. Air monitoring involves inhalable particulate measurements in accordance with AS 3640 [13]. Chemical monitoring methods will quantify the total mass of uranium in a sample, whereas radiological methods will determine the activity concentration, which is used to calculate the committed effective dose. It may be needed as part of assessing risk, as a periodic check on control measure performance, to determine or demonstrate compliance with relevant workplace exposure limits, or where there has been a change in control measures.

VII. CONCLUSION

Although the sampling techniques used to monitor airborne exposures to uranium are similar for both occupational hygiene and radiation protection purposes, the methods used to assess compliance with exposure limits are quite different. As a consequence adherence to one limit may not necessarily mean compliance with the other, and under different exposure scenarios either could dominate. Therefore, it is our recommendation that both the chemical and radiological hazard associated with uranium is assessed, evaluated and controlled.

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Designing resilience – Seismic analysis and civil design of Hinkley Point C heat sink water supply

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I. INTRODUCTION

After nearly 20 years, a new British nuclear power station is being built at Hinkley Point, UK. A significant level of resilience is expected of the nuclear power plant, which is reflected in the UK regulatory requirements. As this takes place post-Fukushima and several other significant seismic events around the world the expectations for resilience in design are heightened. Therefore, a substantial level of resilience is considered for the water-cooling supply system, which is the subject of this paper. As part of the cooling system, the heat sink consists of several marine structures, including the intake and outfall offshore heads. These offshore structures provide the required level of cooling water at any stage of the power plant operation, including extreme events. Therefore, to ensure adequate level of resilience, all plausible eventualities must be considered in the design.

In this paper, the design of six large offshore civil structures, forming an integral part of the Hinkley Point C (HPC) heat sink will be introduced. It discusses the accidental actions, required by the UK regulatory regime, the design process and the construction. This paper will highlight the collaborative approach undertaken within a diverse team consisting of the client (EDF), the designer (Jacobs) and the contractor (Balfour Beatty) and its supply chain, to provide a robust solution that is not only resilient but sustainable for the future.

A. Project Background

A new nuclear power station is being constructed at Hinkley Point, Somerset (UK) by EDF. The new station will generate 3200 MWe of energy utilising two European Pressurised Reactors (EPRs). An important part of the station is the heat sink, which provides the cooling water to the system.

This main cooling water supply for the station is fed from four submerged intake heads located on the seabed approximately 3.4km offshore, in the Bristol channel. These heads are connected to the forebays and the pump house via two 6.0m diameter intake tunnels which carry the water. Once the water has completed its cycle, it is discharged via onshore vertical shafts into a 7.0m diameter outfall tunnel and returned to the Bristol channel via the outfall heads which sit approximately 1.8km offshore. As it stands, there is no system currently in the world that utilises a similar head structure for its head sink and thus there is no precedent to this project.

The functional requirement of the heads is to ensure a critical volume of cooling water is available to the facility at all times, including accidental conditions. These critical structures are designed following the Safety Assessment Principles (SAPs) [2] outlined by the Office of Nuclear Regulation (ONR). A combination of complex analyses, providing the structural demands within the elements, and the capacity design principles, followed by an innovative approach to detailing aiding performance and constructability is presented in this paper.

The product of these efforts is four reinforced concrete intake heads and two outfall heads, complying with the industry standards, including environmental requirements and nuclear safety aspects.

B. UK legislation and the Office for Nuclear regulation

The ONR has established a suite of SAPs, which guide the derivation of regulatory requirements in the assessment of safety cases for nuclear facilities. In order for a facility to attain a UK nuclear site licence to operate it must fulfil the requirements outlined in the ONR's Technical Assessment Guide (TAG) [1] with support of the Safety Assessment Principles guidance [2].

Based on the guidance, as part of the design process, all the internal and external hazards must be considered, including seismic events. It is outlined in [2] that any nuclear classified facility, should be designed to be resilient to a 1 in 10,000 year external event.

The design of the cooling water system is based on a specific general design spectrum for Hinkley Point in the form of Figure 1.

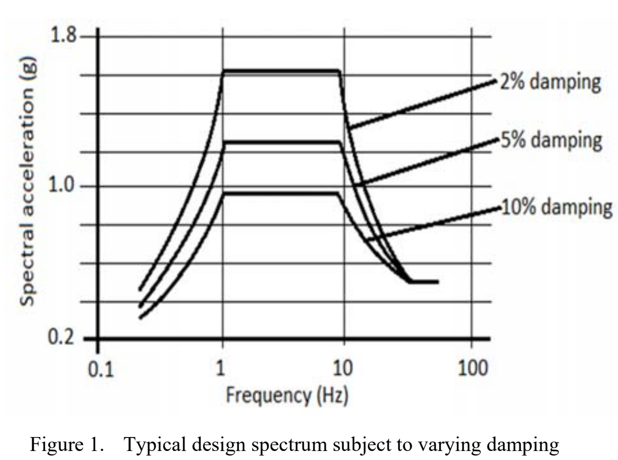

Figure 1. Typical design spectrum subject to varying damping

Jacobs Engineering Group

The structures are located under water; therefore, the water effects are enveloped within the spectra. To confirm that this approach is appropriate, a probabilistic design spectrum based on a Probabilistic Seismic Hazard Assessment (PSHA) was developed for the Hinkley marine structures. This study generated site-specific time histories with values of peak horizontal free field acceleration to the probability of per annum occurrence. During the design process all of these were considered, with the final reinforcement detailing based on the envelope of all cases.

II. Deriving the Seismic Case

A. Background

Seismic ground motions that are not influenced by the presence of a structure are referred to as free-field motions. When a structure is founded on a soil deposit, it can influence the seismic response of the soil and the soil can influence the response of the structure. This process is referred to as seismic soil-structure interaction (SSI).

For nuclear classified civil structures, particular interest is taken in the structural behaviour when subject to earthquake input motion. The dynamic response for a point in a structure is dependent on several parameters, however the following will be discussed in this paper; input motion applied (in all directions), the mechanical properties of the founding soil, the shape and stiffness of the structure and the mass distribution of the structure. When developing a representative SSI model, these parameters must be as accurate as possible.

B. SSI Analysis method

The SSI analysis method follows that of American code of practice ASCE 4, with the use of nuclear industry standard program FLUSH. FLUSH is based on a complex response method with the use of transmitting boundary conditions. The software utilises a 2D plane-strain approach, therefore analysis is undertaken using two perpendicular sections through the structure and underlying soil strata, with properties applied per unit thickness (out of the plane). The nature of the structure, i.e. being symmetrical in those perpendicular directions, allows the use of 2D simplification. There are no 3D effects that affect the response of the structure, other than the consideration of accidental torsion.

The schematic in Figure 2. shows the deconvolution and convolution process that takes place within the soil column analysed in the FLUSH software. The HPC input motion, in the form of time histories, is applied at the top of the free-field within the FLUSH model. The software based on an equivalent linear method predicts the soil degradation, with the resultant seismic motion considered as an input at the base of the FE model with the structure included, allowing the seismic action to act on the structure. This generates a seismic response, allowing the input motion for the structural design to be recorded at the base of the structure.

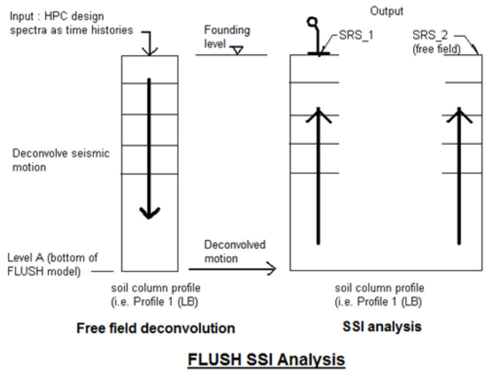

Figure 2. Seismic input motion deconvolution with FLUSH

C. SSI analysis models

The SSI analysis FLUSH models are a representation of the head structure and the specific soil profile. A typical view of a FLUSH model profile is presented in Figure 3. , with varying levels of applied shear wave velocity highlighted. The soil mesh is based on the FLUSH requirement to transfer waves vertically through elements.

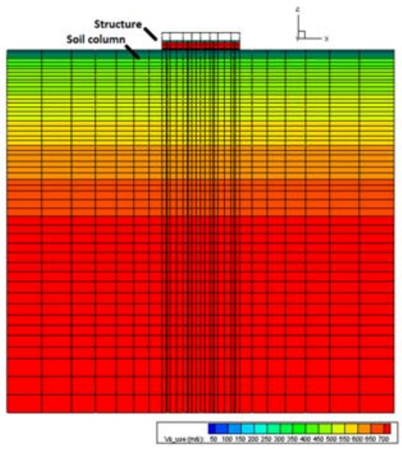

Figure 3. FLUSH soil strucuture interaction model

Using site specific borehole logs the static and dynamic properties of each soil layer are applied to the column from founding level to bedrock. This includes mechanical properties such as: shear modulus, damping levels, as well as their degradation curves, dynamic Poisson's ratio, saturated density and resulting shear wave velocity.

The head structure as shown in Figure 4. is a multipart 3D reinforced concrete structure. Due to the 2D nature of FLUSH software, the 3D head structure is simplified into an equivalent single plane. The structural properties of the very stiff foundation chamber are calculated per unit thickness and transferred into the simplified model. The superstructure of the head is represented by a SDOF stick model, tuned to the principle natural frequency of the superstructure in the direction of motion under consideration.

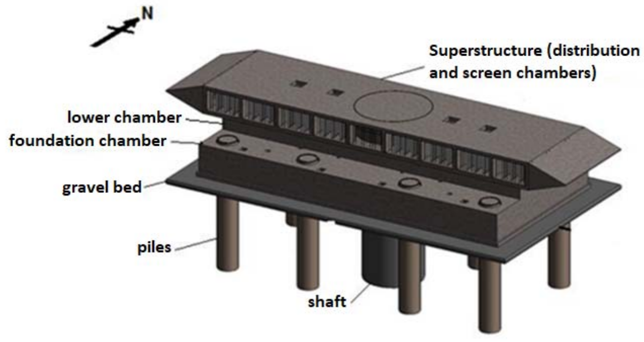

Figure 4. Reinforced concrerte intake head structure

Using the Ritz method [3] in SAP2000, the principle natural frequencies are determined from a fixed base analysis of the full 3D FE model of the head structure. The stick model properties are determined from the reduced Ritz method, whereby dominant frequency information is extracted for the calculation of the equivalent single degree of freedom system (SDOF) in each direction.

D. SSI results

The HPC free field motion is applied as a set of statistically independent time-histories based on a time step of 0.0035s and a transfer frequency range of up to 142Hz, with 0.25g zero period acceleration. The result of the application of this input motion is a suite of seismic response spectra (SRS) for any location of interest within the structure, enveloping all soil and structure conditions.

The SSI study is based on an input with the 1 in 10000 year return period. The broadened and enveloped SRS are then used for the design of the structures, as discussed in the next section.

III. DESIGN PROCESS

A. Finite element

The heads are modelled using FEM software SAP2000. They are formed using an actual desired geometry, using shell elements (Figure 5.). The structure is founded on non-linear springs configured to imitate the expected variability of the soil conditions below. The weight and mass of the head is tuned to represent the actual final weight and mass distribution with all elements included during the construction process.

All static and dynamic actions are applied to the model. Static actions capture permanent, quasi-permanent and frequent actions, such as dead load, soil pressure and hydrostatic pressures. Dynamic actions are broken into two categories; seismic related or unrelated, unrelated actions include such effects as hydrodynamic swell and current.

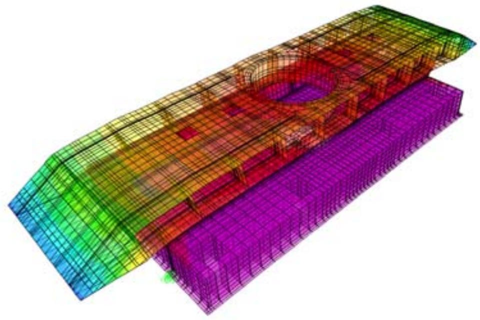

Figure 5. FEM SAP2000 model of intake head structure

The seismic related actions refer to loads such as global torsion, rocking or dynamic response, including hydrodynamic effects. A series of models, with differing restraint conditions, were used to obtain the response of the structure in these different scenarios. All the required actions are combined to build up the Ultimate Limit State (ULS) and Serviceability Limit State (SLS) action combinations utilised for design in accordance with EDF design standard ETC-C.

The HPC design spectra is applied as a response spectrum analysis within SAP2000 in the form of Figure 1. Ritz modal method is used to capture the modal responses, with guarantee of high mass participation capture within the analysis. For all possible actions, results are extracted from each model as a series of unsigned pseudo static forces representing the final/max dynamic response for all directions.

B. Action combinations

To cover all eventualities in design, action combinations are specified for the operational and construction phases. Normal operational action combinations cover the certain and potential situations the head will experience throughout its 85-year design life. The action combinations were rationalised by Jacobs in a combined effort with EDF NNB (New Nuclear Build) and EDF RD (Responsible Designer, France). An agreement of action combinations by all parties required substantial collaborative efforts between the British and French partners. Once agreed the actions are applied and combined into combinations within SAP2000. Actions requiring particular boundary conditions, thermal for example, are combined at a later stage, within the design process.

Due to the nature of dynamic input motion both a positive and a negative effect is possible due to load reversal. For each dynamic action the equal and opposite effect must be considered, and any potential combination of that effect. This leads to a huge number of combination possibilities as each dynamic effect for all directionalities must be combined uniquely, this results in over 700 action combinations covering all possible static and dynamic variations.

All action combinations results are considered within the design process, which consists of a number of calculations that produces reinforcement demands for each group within the structure. Demands are produced for longitudinal and transverse primary and secondary reinforcement and out of plane shear reinforcement. The demands are determined from the modelled shell elements from SAP2000, this is achieved using a sandwich method. The sandwich method takes the in-plane forces and moments of a shell and extrapolates them out to the outer faces of the element based on its thickness. The distribution of the extrapolated forces is influenced by how the shell is stressed, thus if a shell is in pure compression its resultant stress distribution would be different to if it were in pure tension. These extrapolated stresses can then be used to determine the required area of steel to resist the stresses present in each shell. Additionally, consideration is given to the crack widths limits, dictated by the concrete durability requirements.

IV. DESIGNING FOR RESILIENCE

A. Capacity design principles

The importance of gradual and predictable failure modes within any nuclear structure is paramount. Brittle failure modes occur when there is a sudden and unexpected lose of capacity within the section. To ensure that this is designed out of the structure, capacity design principles can be used.

One particular example of severe risk, where brittle mode of failure is not desirable is the lower chamber wall within the head structure. This part of the structure consists of perimeter wall with very limited redundancy in the system. Therefore, there is a significant risk, that if the lower chamber walls were to fail in shear then the shaft hole could potentially be blocked by the collapsed structure, jeopardising the safety functional requirement of the head.

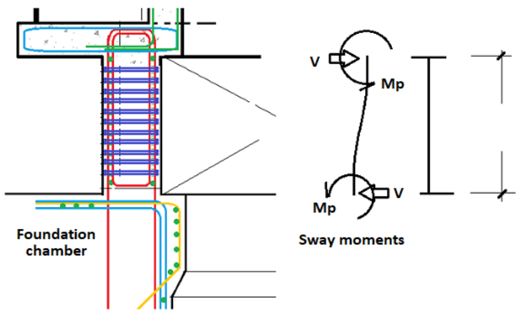

Figure 6. Capacity design detailing of lower chamber walls

A capacity design approach was adopted to ensure ductile behaviour of the wall is observed under overload sway moments in the structure. In simple terms, in order to achieve this, the shear strength of the wall is designed to exceed the plastic strength of the vertical steel. This is to guarantee that the walls will begin to fail in bending before the walls fail in shear. Utilising the ULS provision of steel required in the walls and analysis of sway moments, the shear link provision is fine tuned to exceed the walls strength in bending thus forcing it to exhibit first mode of failure gradually rather than suddenly (see Figure 6.).

B. Beyond the design basis

The ONR require that a suitable seismic margin is provided in order to demonstrate the robustness of the head structures against the design earthquake events. To ensure robustness, an additional case with 1.6 of the design basis earthquake is considered. By applying this seismic margin we can predict the expected modes of failure. The example of the lower chamber wall presented in IV.A., was derived based on such analysis, followed by the capacity design to ensure a ductile mode of failure. Therefore, an adequate seismic margin is present in the structure to perform predictably and fail gradually in bending. Additionally, when possible, the structural detailing is revised to design out brittle modes of failure.

V. DETAILING

Due to the balance of different requirements, involving the flow of water, structural robustness and durability, a complex arrangement of reinforcement and embedded items is required. As a result, a specialist 3D modelling approach was utilised to produce a clear reinforcement layout and undertake a full clash detection. Each bar is added to the model in a certain order and is uniquely tagged for quality processes, the order is selected to ensure means of access and buildability when passed over to the contractor. A collaborative approach, involving all parties early in the process, is taken to ensure full confirmation that the drawings meet the design intent, but are also buildable. This was achieved through periodic workshops, which have proved vital in ensuring that the approved design is delivered to the contractor while maintaining UK nuclear quality standards.

VI. CONCLUSION

Resilient design and nuclear safety culture is about expecting the unexpected. It is about being prepared for every eventuality and ensuring preventative measures are in place to maintain confidence in your system. It is about accurately modelling some of the most complex actions and adequately detailing your structure to perform to the required standard.

This paper introduces the analysis and design methodology of an important civil nuclear structure. Due to the risk associated with failure of this system it has been designed to be resilient, even beyond its design basis. This paper delivers an insight into the level of collaborative effort that is required between all involved parties for attaining this level of resilience. It is only through cooperation that this is attainable.

VII. ACKNOWLEDGMENT

Thanks to Konrad Kukla of Jacobs, for his endless efforts to help me develop my understanding and career.

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The Feasibility of Blockchain Application to Nuclear Safeguards in the IAEA

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I. Introduction

Nuclear safeguards, first established under the International Atomic Energy Agency (IAEA) Statute in 1957, are measures designed to ensure safeguarded items, such as nuclear material, "are not used in such a way as to further any military purposes,"[1] and have been critical to nuclear related international and domestic politics since the 1950s. Furthermore, accounting for nuclear materials has expanded overtime to include a wide range of organizations, technologies, and collaboration to prevent the further weaponization of nuclear material.

One emerging technology that could revolutionize how the IAEA oversees its safeguards protocols is blockchain. Blockchain, first introduced by Satoshi Nakamoto in 2009,[2] has potential applications outside economics that have yet to be fully realized. With further research, clearer insight can be gained on the relation between blockchain and more accurate safeguards accountancy. Blockchain has the capability to revolutionize nuclear safeguards, but first its feasibility must be evaluated.

Blockchain is a necessary next step towards ensuring nuclear security because of its implications towards real time reporting of material accountability. Currently, transit matching protocols can take weeks to months to be fully reported. With the ever-present threat for nuclear material diversion, it is imperative that potential diversions can be instantly flagged, and this capability can be achieved via blockchain.

The overarching question guiding the research is, could blockchain be used as an additional layer of verification to improve current nuclear safeguards protocols in material accountability under the IAEA? Sub-research questions are, would the application of blockchain be technically, legally, and politically feasible? What are the challenges to blockchain integration? And what form would blockchain take if incorporated into IAEA nuclear safeguards?

II. Blockchain Formats

The most well-known blockchain structure is the bitcoin blockchain which is a permissionless public ledger, meaning anyone can join, there are multiple nodes maintaining the ledger, and there is no central authority on the system [3]. However, there are other blockchain structures. These various structures can be used as a conceptual framework to assess the feasibility and applicability of blockchain in various industries.

One potential variety to the bitcoin blockchain is a permissioned system. "Whereas the bitcoin blockchain is entirely open and *permissionless*–that is, anyone can access it and interact with it–*permissioned* blockchains require users to have certain credentials, giving them a license to operate on that particular blockchain." [4] Permissioned blockchains have application for use between or within organizations looking to integrate blockchain. Access to a permissioned blockchain would be at the discretion of the organization using the blockchain and would prevent individuals with no relevance to the blockchain's objectives from accessing it.

A. Public Ledgers

In addition to being permissioned or permissionless, the blockchain can have different ledger structures. The most well-known is the public ledger, which the bitcoin blockchain utilizes [5]. A public ledger is distributed and decentralized meaning all participants on the blockchain have equal access to viewing, validating, and mining the ledger [5]. A permissionless public ledger would mirror the bitcoin blockchain; however, in using the IAEA, under a permissioned public ledger structure only IAEA member states would have access to the ledger, and all would play an equal role in mining the ledger. A public ledger would be most appropriate in industries where there is no trusted third party for individuals to interact through; therefore, public ledgers hold little relevance to nuclear safeguards under the IAEA, whom are a trusted third party.

B. Private Ledgers

Private ledgers are centralized and localized meaning there is one ledger mined by a single miner, however, other miners can view the ledger [5]. Following the IAEA example, under a private ledger only the IAEA would mine the ledger. A permissioned private ledger could mean only the IAEA can view the data being submitted whereas a permissionless private ledger could mean any member state can view data on the ledger but cannot help mine the ledger. A private ledger would be the most likely form if the organization were to integrate blockchain without changing its hierarchical structure; however, a private ledger is very similar to a database and may not add additional value to the safeguards structure.

C. Consortium Ledgers

The final type of ledger is a consortium ledger which is centralized and distributed and requires a fixed group of participants to mine the ledger [5].Again, under the IAEA example, a consortium ledger could be permissioned and require the IAEA to select the member states operating on the ledger or permissionless and require those mining the ledger to be randomly selected. A consortium ledger can combine the low trust established in a public ledger with the trusted third-party required of private ledgers [6]. A consortium ledger would be the most likely form if the other two ledger formats were considered unworkable.

III. ESTABLISHING KEY IAEA PILLARS

A. Technology

To oversee its safeguards requirements, the IAEA incorporates innovative technologies into their safeguard's protocols. Starting in 1966, the IAEA held symposiums on nuclear material safeguards as an avenue for the exchange of ideas on technologies related to nuclear safeguards [7].

Starting in the 1980s, the incorporation of new technologies began to accelerate [8]. Some of these technologies include the use of satellites to quickly send data back to IAEA headquarters [8]. Another was the Cherenkov viewing device, which detects light emitted by spent fuel underwater [9].

Recently, the IAEA added an unmanned surface vehicle that floats atop spent nuclear fuel ponds to detect whether nuclear material is being stored [9]. Technology is an imperative part of nuclear safeguards oversight; therefore, the IAEA constantly seeks new ways technology can aid nuclear safeguards management.

B. Law

The IAEA Statute lays out the framework for how nuclear safeguards are established and implemented. Article III.A.5 establishes the IAEA's authority to create safeguards, Article XII establishes inspectors who verify compliance, and Article XII.C establishes what measures these inspectors have access to [10]. While the IAEA Statute develops the responsibilities and legal reach of the IAEA, there are other legal documents that aid the development of and adherence to IAEA safeguards.

One type of legal document that plays a major role in establishing what states fall under the jurisdiction of IAEA safeguards is multi-lateral treaties. A major development in the reach of the IAEA was the Treaty on the Non-Proliferation of Nuclear Weapons (NPT), specifically Article III [11]. IAEA INFCIRC/153(Corr.) was soon established after laying the groundwork for the application of nuclear safeguards under the NPT [12]. Additionally, nuclear weapons free zone treaties discuss the application of IAEA safeguards to their region, [13] which reaffirms the regional commitment to IAEA safeguards. Law is a central guideline on what safeguards are and who they apply to making it a central pillar to the IAEA.

C. Politics

The establishment of the IAEA was guided by states' political interest. The IAEA was established in 1957 out of fear of the direction nuclear technologies could take [14]. The organization grew out of U.S. President Eisenhower's 1953 "Atoms for Peace" speech which came five years after the U.S. tested its first thermonuclear weapon and two years after the USSR tested their first thermonuclear weapon [15]. Not only was the IAEA established for political reasons, but politics continue to play a role in what states accept various levels of oversight.

IV. CHALLENGES TO INTEGRATING NEW TECHNOLOGY

A. Technological Perspective

Blockchain "will not radically transform the safeguards information ecosystem, but it will allow operations to be refined and adapted to an evolving safeguards system." [16] Nuclear safeguards systems are technology oriented, continuously seeking the next advancement to further the safeguards objectives. Smooth technological integration is necessary for convincing states blockchain is the next technical advancement for nuclear safeguards. Blockchain provides smooth technical integration because it is an advancement of present database technology. Keeping technical integration simple is necessary for the advancement of safeguards technology because the current database software works well enough. Integration of blockchain only comes if it can be proven that integration is not cost extensive in time, money, or resources and can correct faults in the current system.

B. Legal Perspective

The main legal challenge the application of blockchain faces lies in INFCIRC/153(Corr.). Under part one of INFCIRC/153(Corr.) the document outlines the establishment of national systems for accounting for and control of nuclear material. Paragraph 7 of INFCIRC/153(Corr.) states:

> The Agreement should provide that the State shall establish and maintain a system of accounting for and control of all nuclear material subject to safeguards under the Agreement, and that such safeguards shall be applied in such a manner as to enable the Agency to verify, in

ascertaining that there has been no diversion of nuclear material from peaceful uses to nuclear weapons or other nuclear explosive devices, findings of the State's system. [17]

While further in the document there are guidelines for what the state's system must measure and monitor, [18] ultimately it is the state's decision as to the form of its accountancy system. INFCIRC/153(Corr.), therefore, becomes an issue for blockchain's application in the IAEA. While the IAEA could suggest to their member states a new system be implemented, under INFCIRC/153(Corr.) it is the state's decision whether they utilize the new accountancy system.

Therefore, the best way to incorporate blockchain comes to convincing states it is in their best interest to use a new technology or create a new legal framework that specifically requires blockchain to be used or grants the IAEA permission to determine what system format to use for accountancy.

C. *Political Perspective*

While the IAEA implements technical means to carry out its safeguard duties which are imbedded into the system via legalities, the IAEA is first and foremost a political organization. Politics can influence the regulatory capabilities of an organization [19]. For example, in the IAEA states either agree to or disagree to oversight by the IAEA depending on their political interest because states always act in ways that advance their own interests.

Therefore, states need to be shown current software is not operating in their best interest because vulnerabilities exist potentially allowing for covert nuclear program development. By emphasizing the dangers of covert nuclear programs and how blockchain can potentially minimize nuclear material diversion risks, states can begin to see why blockchain is such a necessary technological advancement.

To incorporate blockchain into nuclear safeguards protocols it becomes about convincing states their best interest lies with blockchain. In showing states that blockchain is in their best interest, it could alleviate other obstacles blockchain faces such as Paragraph 7 of INFCIRC/153(Corr.). Through understanding the self-centered nature of politics, one can begin to discern the political challenges that arise in attempting to implement blockchain into IAEA nuclear safeguards.

V. BLOCKCHAIN INTEGRATION IN IAEA SAFEGUARDS

While theoretically a public system that allows for all states to act as miners on the blockchain would alleviate some of the challenges brought forth by current material accountancy practices, realistically there are hurdles to implementing a public system. The challenge to having universal miners for data access and entry is Paragraph 7 of INFCIRC/153 (Corr.). To wait for universal acceptance of blockchain among member states, could mean waiting decades before the benefits of real time material accountability is achieved.

Therefore, a consortium system, which only requires a handful of states to act as miners, is the most likely way blockchain could be integrated into the IAEA. A consortium system does not require all states to participate as miners or the organization to adopt the technology, but instead can be applied to specific nuclear accountancy scenarios. An example would be the Akkuyu nuclear facility in Turkey [20]. The Akkuyu nuclear facility is a Russian owned nuclear power plant on Turkish soil [20]. Because two states are involved both must report to the IAEA on this singular facility.

Instead of having both Russia and Turkey reporting separately to the IAEA, a consortium could be used between Russia, Turkey, and the IAEA, where all three entities act as miners, to eliminate the IAEA from receiving duplicate information [20]. Since specific use cases exist for consortium ledgers there is greater political likelihood of success because only a handful of states would need to act as miners and adopt the technology.

Consortium ledgers can be used to minimize the overlap that occurs between agency accountancy by consolidating the information gathered by various nuclear safeguards bodies into a single space. A consortium ledger overcomes the legal barriers created within INFCIRC/153(Corr.) because it is established through unique legal documents pertaining to the specific situation that will be put on the blockchain. Politically, a consortium ledger for specific use cases is more likely to be achieved because it may only require a few states/entities to agree to become miners. Even then, the ledger would only be used in the specific scenario it is seeking to monitor, not the entirety of the accountancy system.

VI. CONCLUSION

The feasibility of blockchain in IAEA nuclear safeguards was assessed by looking at the integration required between technological, legal, and political components of the IAEA before assessing which of the three ledgers presented the greatest feasibility for integration. The analysis indicated a consortium ledger presents the greatest feasibility for integration because it realistically addressed the challenges under the three IAEA pillars. Technically, a consortium is differentiated from a database in its role of allowing multiple bodies to mine the ledger. Legally, it can circumvent INFCIRC/153(Corr.) because it would require legal documentation for specific use cases. And politically, it only requires agreement between states in the specific scenario where blockchain is being utilized.

The current challenges to blockchain integration were presented by assessing the three pillars within the IAEA. A challenge to technology was discovered to be proving that blockchain would be more efficient than a standard database. The challenge to law is working with and around

INFCIRC/153(Corr.) to understand how blockchain integration can be legally allowed. Finally, a challenge to politics is determining how to show states blockchain integration is in their best interest.

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A Collaborative Experience: Implementation of an Optimised Tc-99m Generator Assembly Process

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I. Introduction

Technetium-99m (Tc-99m) is a widely used radioisotope in diagnostic nuclear medicine. It is a favorable radiotracer that can be chemically incorporated into drugs targeting specific tissue types which can be imaged due to the 141keV gamma emission energy [1]. Its six-hour half-life makes longer transportation times a challenge [1, 2]. This is overcome by delivering the parent isotope Molybdenum-99 (Mo-99) in the form of a generator. Mo-99 has a longer half-life of sixty-six hours allowing for a calibrated activity of Tc-99m to be milked from the generator for immediate use upon receival [2].

Currently Australia's Nuclear Science and Technology Organisation (ANSTO) is one of the largest manufacturers and suppliers of Tc-99m generators in Australia. Mo-99 is produced at ANSTO's Australian Nuclear Medicine (ANM) facility. This is further processed at ANSTO Health Products and delivered to customers within shielded GENTECH™ generators. The GENTECH generator production method is a justified process from which patients all around Australia benefit. This paper aims to demonstrate the implementation process undertaken to review, identify and mitigate risks to achieve occupational dose optimisation.

II. Methodology

A. Augmented Risk Analysis

As part of ANSTO's questioning attitude, evolving safety culture and application of best radiation protection practice [3, 4], an augmented risk assessment was used to update the risk of potential skin exposure in the generator assembly process. This was initiated by collaborative discussion between production facility subject matter experts, engineering group and radiation protection services (RPS) – and was overall achieved through a series of steps. These steps included process familiarity, identifying potential exposures and initial monitoring observations.

1) Process familiarity

Collaborative discussion between production facility subject matter experts, engineering, and RPS was vital in gaining a representative risk assessment of the generator assembly process. It is important to understand why certain methods are followed to reduce adverse effects as a result of implementing change. Standard operating procedures were studied, process video recordings were analysed and radiological monitoring of the process in the cleanroom was performed. The assembly process consists of several assembly steps executed at designated workstations by different personnel.

2) Potential Exposures Identified

Table I identifies the risk outcomes from the professional review process. Contamination levels observed from the technical radiological monitoring in the cleanroom highlighted a step in the assembly process occurring at workstation 1.

TABLE I. Risks Identified in Current Process

Activity with Significant Risk	**Risk Scenario**	**Mitigation Strategy**
Transfer of shielded generator pot	Contact with unshielded generator vial or to the contents of vial if partially assembled generator pot is dropped	• Engineer method to secure pot lid during transfer • Modify lifting cradle • Reinforce incident response procedures
Manual handling during generator assembly	Contamination from handling of generator leads	• Additional contamination monitoring in process: identify and act • Frequent glove changes to reduce exposure time • Alternate assembly methodology using tools - distance from potentially contaminated generator components
Transfer of shielded elution pot	Drop of elution sample in pot	• Vial redesign • Pot redesign • New generator process
Manual handling of elution	Handling errors leading to contamination	• Additional contamination monitoring in the process: identify and act • Frequent glove changes to reduce exposure time

At workstation 1 a partially constructed generator containing a Mo-99 column with exposed elution lines is received. Dust caps protecting the elution lines are removed and replaced with different apparatus. Consistent re-occurrence of gross contamination was observed on operator gloves post task, suggesting transfer of contamination from the dust caps.

3) Monitoring and Observations

Dust caps were collected and sent for gamma spectroscopy by the waste characterisation group. Between 0.0268 kilobecquerels and 54500 kilobecquerels of Mo-99 and Tc-99m was identified.

The activities contained in the generators at time of assembly would range from 20 to 476 gigabecquerels and therefore potential dose received from skin exposure was identified as a risk. RPS professionals used simulation to calculate possible doses from the dust cap activity results in conjunction with estimated exposure times, geometry, transfer factor and contamination skin dose factors derived from ICRP 119 [6]. These factors have been quoted in literature from many other re-modelling structures [7, 8]. The outcome identified scenarios that could result in workers receiving dose above the threshold which scientific literature explains begins an onset of skin tissue reactions e.g. transient erythema, blistering [9, 10, 11].

B. Process Change

The potential outcome was discussed amongst the collaborative review group to suggest appropriate and realistic solutions to mitigate this risk. Factors such as: worker safety, facility resources, limitations and ability to continuously support product manufacture and supply chain were considered.

TABLE II. SUMMARY OF NEW PROCESS CONTROL

Control type	Improvement	Pre-implementation
Substitution	• Introduction of tools: protect operators from high contamination levels and associated dose. • Minimise risk of contamination transfer onto operator gloves and hence potentially operators' skin.	• Mock trials for usability and desired functions (hold a line and remove a dust cap). • Cleanroom validation for use in TGA pharmaceutical grade GMP production facility.
Administrative and PPE	• Teach and train self-glove monitoring techniques in elevated dose area. • Improve safety culture. • Teach and train workplace monitoring for contamination spread and dose rates.	• Teaching plan designed and discussed. Including: • Suitable placement of monitors • Proper glove check techniques • Frequent glove changes incorporated into work instruction.

Table II displays two possible control types discussed for implementation. The tool control was trialed, timed and implemented in a mock cell. Production operators of varying levels of experience were trained and video recorded undertaking the new process with feedback integrated into the improved process flow. RPS analysed the video recordings to generate approximate timings for dose assessments. The Administrative/PPE control was discussed prior to monitoring to ensure all production workers were aware and understood the change.

C. Continuous Monitoring and Optimisation.

Monitoring was conducted for an initial duration of seven weeks accounting for initial unexpected delays, tool training, tool familiarity, human factors [12] and to capture changing conditions during production ramp up. A follow up monitoring campaign was conducted 3 months later for two weeks.

During this time levels of radioactive contamination and dose incurred from different assembly steps was monitored, results from finger and wrist TLD's issued for the program were recorded and dust caps were collected for contamination activity quantification via gamma spectroscopy.

RPS utilised these inputs to generate a new training program tailored to the operators. The training program incorporated radiation protection during generator manufacture and provided reassurance about the levels of exposure.

III. RESULTS AND DISCUSSION

Figure 1 shows the immediate reduction of contamination on operator gloves since the introduction of tools to the process. Overall the occurrence of contamination has reduced from 100% in the first run without tools to 6% occurrences from a total of fourteen runs observed with tools.

Table III shows the best estimate for potential equivalent dose averted to skin for an average worker per generator and per annum with the optimised process.

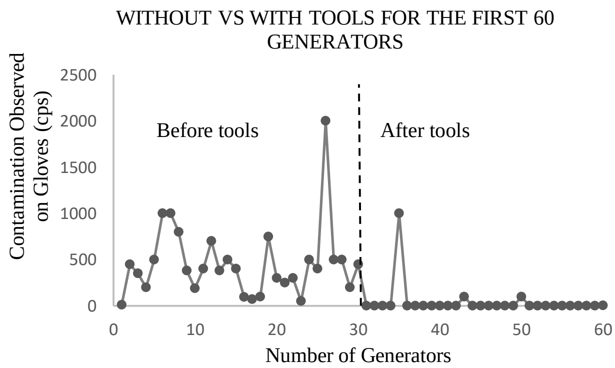

Figure 1. Graph of observed contamination between the first run without tools and second run with tools.

TABLE III. POTENTIAL AVERAGE DOSE EQUIVALENT AVERTED TO SKIN RECEIVED

Control type	Potential Average Dose Equivalent Averted to Skin Received (mSv)
Per generator (with tools)	0.3
Annually (with tools)	103

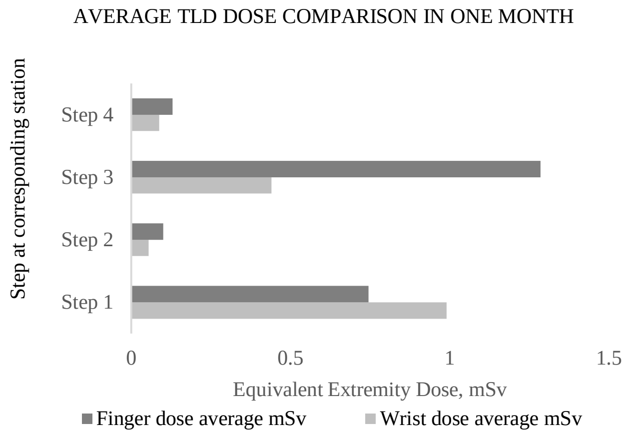

Figure 2. One month average TLD dose comparison between different steps.

This was modelled from the dust cap activity analysis results by the professional radiation protection advisors using VARSKIN 5.3 [5, 6]. The initial aim of the dose-reconstruction was to provide representative estimates of the potential exposure for the group of workers involved. This allows further explanation and contextualisation of what that exposure means in terms of radiological risks.

Limitations and challenges in this study included the exclusiveness of the TLD data. Even though the TLD's were exclusively issued to the production team for the monitoring program, they weren't exclusively controlled to each step. Therefore, extremity dose analysis couldn't be statistically differentiated from each step in the assembly process. However, through control documentation records TLD data extrapolated in figure 2 was generated by selecting individuals that had worked at the designated workstation twice consecutively in one month.

Step 1 located at workstation 1, the identified as an area for improvement still has the higher wrist dose. This comparison also highlighted highest finger dose received while working in station 3 – this is most likely due to closer proximity, more hand related work steps and a longer task duration. This learning experience highlighted a future dose optimisation opportunity for step 3 and an improved dosimetry data gathering and analysis technique for a future review. It should be noted however the risk of skin contamination exposure at workstation 3 is approximated less than that of workstation 1. Only 39% of total glove contamination occurrences was observed at workstation 3 (in which 17% $\geqslant$ 50cps) as opposed to 57% observed at workstation 1 (in which 72% $\geqslant$50cps).

Reflection and experience of this study demonstrates that dose optimisation would not have been successful without:

- A systematic data collection methodology to establish potential exposure levels before and after the implementation of mitigation strategies.
- A slow-down of throughput of the process as manufacture re-commenced allowing operators to learn and adapt the new techniques of the safety measures introduced.
- Ensuring the risks are communicated and understood by all.
- Open communication and feedback regarding protection and safety with all stakeholders.
- Concurrent work to support changes affecting product GMP and compliance.

IV. CONCLUSIONS

In conclusion the implementation of new controls and mitigation strategies proved effective in reducing the skin dose risk. Exposures can be deemed optimised for the prevailing circumstances since radiological monitoring data and statistical modelling showed the potential skin dose averted by implementing these controls and mitigation strategies. Review and optimisation continues with a feedback loop for ongoing process improvements.

ACKNOWLEDGMENT

This review and implementation process would not have been possible without the collaborative group consisting of the Health Products team, waste characterization team, engineering team and RPS. The scope of this study proposed by the senior radiation protection advisor and health physicists is large and much work in data analysis, remodeling and optimization was performed by this team. Thank you all.

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Understanding Digital Trust and Segmentation in Nuclear Facilities

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I. Abstract

A comprehensive understanding of a network is vital to providing computer security risk management, especially in the context of nuclear security. With rapid changes in the IT and OT landscapes, new technologies and methodologies are changing the way we use computer systems every day. In a nuclear security environment, we need a way to maintain visibility of all systems and services deployed on our networks and their interactions with each other.

We are developing a tool that maps out the network landscape, including all of the trust relationships that devices have across networks and security levels, with the end goal being to simulate an adversary attacking various points of the network and thereby understand the extent to which malicious actions can be carried out.

In doing so, we are gaining a better understanding of trust relationships on a network and employing security controls in order to reduce the likelihood of an adversary obtaining control of systems at a large scale.

II. Introduction

This paper describes the tool that we have developed to promote understanding of the importance of increased security controls through security layers in a nuclear space. The tool is primarily focused on how an adversary can pivot through network to access more secure nodes by using vulnerable nodes, in accordance with the concepts proposed by M. Hewes [1].

It is natural for the IT and OT landscape to grow due to changes and innovations leading to new technologies and methodologies influencing the way we use and setup computer systems. For this reason, it becomes difficult to provide adequate risk management on a network that is constantly evolving.

An implicit detail is that various systems have methods of trust between other systems. When considering that these trust relationships can traverse multiple security levels, it becomes clear that certain systems can reach across to a number of other systems using these trust mechanisms. If an element within a network (such as a web server in the border or a workstation within a corporate network) were to be compromised by an adversary, their reach could extend not only to other systems, but also into potentially distant security levels.

Due to the evolving nature of systems and their architecture and the complexity of large networks, understanding potential downfalls in the design of a network is a cumbersome task. Furthermore, taking into account varying attack vectors and using them to understand the extent to which an adversary can reach is difficult.

Our proposed solution is to develop a training tool that allows users to map out security levels, devices and the trust relationships that tie them together. This tool can then be used to utilise the defined network and trust relationships to simulate adversarial attacks at multiple entry points and report on the attack vectors that can be abused over the reach of trust relationships. This information can then be used to train users and provide awareness of different forms of network traversal attacks and how an adversary can traverse through security levels by abusing trust between systems.

III. Mats/methods, results, discussions

The decision was made to build this tool as a web application in order to operate within modern browsers on most operating systems. In order to achieve this, we used the following frameworks and tools:

The tool will be primarily be used for training, therefore it should be intuitive so that the time spent on the tool is not spent learning how to populate the data but is instead spent understanding the concepts represented by the tool. This is especially the case when dealing with complex network diagrams.

d3.js [2] is an open source JavaScript library that was chosen to draw the graph and provides the ability to interact with elements on the screen. It allows us to plot all of the data provided from the backend in a way that lets the user clearly understand the complex network and relationships. The library provides the ability to drag, move and modify elements on the screen in order to tailor the layout of the data to the user.

Trainers may want to the ability to create scenarios in advance as training material, users may want the ability resume editing their diagram and the application should provide a mechanism for restoring work from a save state should the user experience an error. A form of data persistence would achieve these requirements.

The Symfony [3] PHP framework is used to drive the backend. This powerful set of libraries allows us to build an MVC style application that is customizable to our needs. In order to achieve data persistence, we utilised an Oracle database to allow us to persist and recall information provided by a user.

Using the application, we can construct a mock network with security levels, zones, systems, subsystems and define the relationships between them. Users can define zones and elements and place them on a canvas. Zones can be placed within a predefined grid of security levels and elements can be placed within zones, allowing the user to indicate the hierarchy that these zones and elements have with each other. Furthermore, elements can be created with parent-child relationships allowing further levels of inheritance to be established on the canvas.

For all elements, trust relationships can be defined between them, outlining the direction of the relationship, security measures that take place over the relationship and a list of actions that can be acted upon utilizing the relationship.

Based on all the provided information and careful creation of the network's definitions and parameters, simulations can be run against the mock network.

The end result is a diagram with elements and trust relationships highlighted in red, denoting the compromised components of the network and how they were affected. These results can be used to help strengthen any weak points in a network to aid and assist in preventing any susceptible components of a network to an attack in the likes of the simulation.

IV. USAGE

Users can access the application and begin defining zones and elements, providing relevant information when necessary to help denote various components of a network. Icons and elements are placed on the screen and can be dragged around and positioned in appropriate locations around the canvas. Elements can be placed within zones, and zones can be placed within security levels. Trust relationships can then be created by linking two elements together and providing details about the connection such as the services used.

After the user builds a diagram, it can be used to simulate adversarial attacks. A user can select an element and flagging it as adversarial. This indicates that the element has been compromised and is being used to reach out further into the network. Utilising the parameters defined trust relationships that interact with the element, the simulation takes into account available actions and security measures and denotes if an adversary could potentially reach the next element. For each successful hop between connected elements, the same logic is applied based on the parameters of the new element, traversing through the network as far as possible based on the parameters of any connecting trust relationships.

In order to aid the simulation, trust relationships make use of MITRE ATT&CK [4] vulnerability library to define how a trust relationship is built between two computer-based elements. By incorporating a database of known attack vectors, we can approximate how a relationship can be compromised by an adversary so that we can drive the simulation.

V. EXAMPLES

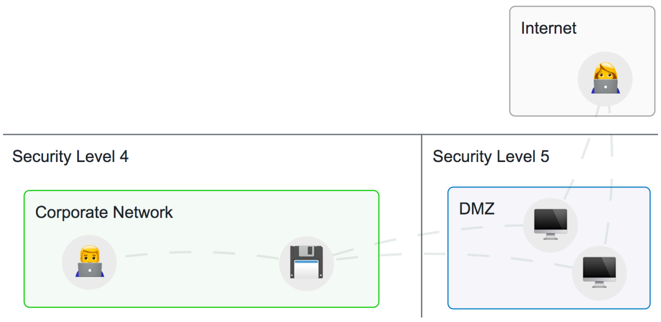

Figure 1. An example of a network defined within the application

Figure 1 describes the following scenario. Two web servers (development and production) reside within the DMZ. These servers have a connection to a database within the corporate network, which trusts the web servers. There is a database administrator within the corporate network who is trusted by the database and an external user on the internet who has access to the web servers in the DMZ. Trust relationships have appropriate services defined.

Using this example network, we can begin probing various points of failures and ways to traverse throughout the network. For example, flagging the user in the internet as an adversary will begin using the trust relationships they share with the production and development web servers in the DMZ to find a weakness in order to pivot to the database server within the corporate network. Similarly, we can test mark the database administrator as an adversary, should the administrator be an insider threat, based on their granted permissions and the systems they have direct access to, we can determine how far into a network they can penetrate by all means necessary.

Based on the output of the simulation, the information provided is aimed to help make informed decisions to help strengthen a network and limit how far an adversary can reach across a network. Since persistence tactics [5] are commonly utilised, an adversary can continue to attempt to use a vulnerable node to further penetrate into a network. Understanding how a node can be used to further penetrate into a network can provide insight as to how an adversary can

VI. Conclusion

In most cases, trust is imperfectly implied. Adversaries can utilise these trust relationships to undertake malicious actions and compromise relationships held with other elements. By abusing multiple relationships, an adversary has the ability to traverse across multiple security levels, compromising systems and processes along the way. While in theory it is a simple attack, it is difficult to predict as the IT and OT landscape continues to grow evolve.

Providing a training tool to help comprehensively visualise a network and its related components while also providing simulated traversal attacks can assist significantly with training in computer security risk management.

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Study of Radioactive Corrosion Products in Dukovany Nuclear Power Plant at Czech Republic

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I. Introduction

Radioactivity is a natural phenomenon and natural sources of radiation are features of the environment. Radiation and radioactive substances have many beneficial applications (medicine, industry or agriculture) but radiation risks to workers and the public and to the environment may arise from these applications and must be assessed and controlled. The operation of nuclear power plants is one of the sources of radiation fields. According to the international safety standards and principles such as ALARA principle one of the major tasks of Dukovany Nuclear Power Plant (type VVER-440) is to minimize occupational radiation exposure of plant personnel involved in operational tasks such as maintenance, inspection and repair mainly during outages for refueling [1], [2].

Occupational radiation exposure and radiation field generation are based on abundance of radioactive corrosion products (RCP) in primary circuit. These radioactive corrosion products may be generated directly, by neutron activation of structural materials (Nickel and Cobalt) of reactor core under high temperatures, or indirectly by activation of corrosion products from out-of-core surfaces released into the primary water coolant and transported by primary coolant either as soluble species or colloidal/inertial particulate species [2], [4]. These products can deposit in form of crud on fuel rod surfaces and may induce axial offset anomaly (AOA) of the fuel rods [3].

The primary water chemistry is based on reducing alkaline conditions with constant high-temperature pH_{300} and the presence of specific concentration of boric acid, ammonia/hydrazine and potassium. This primary coolant might be corrosive for the structural materials of primary technology and therefore is also one of the key factors for RCP formation [3], [6].

Another factor, which may influence formation, composition and abundance of radioactive corrosion products is operational process of reactor system with changes in nominal power of nuclear reactor, either planned or unplanned, the transients (process of shutting down) and the duration and workflow of planned and unplanned outages. For example – the oxygen access to the primary components during the outages might influence the process of corrosion on the surface of primary structural materials [4], [6].

The radiation fields issue is the base for the assessment of the whole radiation situation in the nuclear power plant, which is provided by continual monitoring of radioactivity in operational media, in the whole technology and workplace. The occupational radiation exposure of workers is monitored in the whole area of the NPP. The Group of Chemistry is involved in this important task by continual monitoring of media in all three cooling circuits, by periodical sampling in specific sampling locations in technology, by in-situ gamma spectrometry of primary surfaces and by sampling of RCPs in the primary circuit during the outages [5], [6].

II. Analysis of Radioactive Corrosion Products in Dukovany Nuclear Power Plant

A. RCP in Dukovany NPP

The structural material of primary circuit in Dukovany NPP (steam generators, reactor, primary tubes and other components) it made of austenitic stainless steel 08Ch18N10T. As far as we know, the dominating radioactive corrosion products are ^{58}Co and ^{60}Co, which are generated in austenitic stainless steel by thermal neutron reaction from ^{58}Ni (Table I) and which are the major contributors to the personnel dose. In addition, significant amounts of ^{51}Cr, ^{54}Mn and ^{59}Fe are present, although these radionuclides are minor contributors to the radiation fields, and small amounts of ^{124}Sb, ^{95}Nb, ^{95}Zr and ^{125}Sb are detectable on surfaces and in the coolant. The Dukovany NPP has had an excellent corrosion product behaviour, which produces low primary circuit radiation fields, based on low RCP surface activities. Considering that nuclear fuel replacement (with higher ^{235}U enrichment) has been currently in process, the changes in the steady trend of RCP behaviour (new monitored radionuclides) are expected. The major new contributor is ^{125}Sb. The other new detected radionuclides are ^{65}Zn, ^{103}Ru, ^{125}Sb, ^{154}Eu, ^{156}Eu and ^{181}Hf [3], [6].

The following Table I. contents the summary of the most significant radioactive corrosion products in Dukovany NPP [3].

TABLE I MAJOR RADIOACTIVE CORROSION PRODUCTS IN HOT AND COLD LEGS OF STEAM GENERATORS IN DUKOVANY NPP [3]

Radioactive corrosion product in HL and CL of SG	*Half-life*	*Source*
^{51}Cr	28 days	^{50}Cr, ^{54}Fe
^{58}Co	71 days	^{58}Ni
^{60}Co	5.3 years	^{59}Co, ^{60}Ni
^{124}Sb	61 days	^{123}Sb
^{95}Nb	35 days	^{95}Zr
^{54}Mn	312 days	^{54}Fe
^{59}Fe	45 days	^{58}Fe, ^{62}Ni
^{125}Sb	2.7 years	^{58}Ni

B. Methods of RCP evaluation in Dukovany NPP

The periodical monitoring and evaluation of radiation situation during the outages by has been provided by external supplier. The data results from two different methods. The first one is the In-situ gamma spectrometry, which is performed on inner surface of primary tubes (hot and cold legs of steam generators). The second one is an electrochemical sampling of corrosion crud in hot and cold collectors of chosen steam generator. In-situ gamma spectrometry has been provided by special measurement apparatus based on semiconductor detector with germanium crystals. The measured data are evaluated in specific Gamwin software and gives the results in form of specific surface activities of radionuclides on inner surface of primary circuit. The second method gives results based on gamma spectrometry measurement of radionuclides surface activities in samples.

III. DISCUSSION

TABLE II COMPARISON OF DATA FROM TWO DIFFERENT METHODS (IN-SITU GAMMA SPECTROMETRY AND ELECTROCHEMICAL SAMPLING)

Radioactive corrosion product	*Surface activity [kBq/cm²] - Electrochemical sampling*		*Surface activity [kBq/cm²] - In-situ gamma spectrometry*	
	Hot leg	*Cold leg*	*Hot leg*	*Cold leg*
^{54}Mn	1.75	0.14	1.58	0.698
^{58}Co	5.70	0.53	8.47	2.51
^{59}Fe	0.22	0.03	0.747	0.579
^{60}Co	2.84	0.35	5.65	3.07
^{95}Nb	1.49	0.28	3.9	1.23
^{110m}Ag	0.18	0.99	0.246	0.262
^{124}Sb	1.98	0.07	2.75	0.754

The TABLE II shows the results of the two different measuring methods – the In-situ gamma spectrometry and the electrochemical sampling of corrosion crud. For the analysis of radiation situation trends are used the data of surface activities (As) based on radiochemical measurement of corrosion crud from hot and cold collector of SG35 and the data of As based on In-situ gamma spectrometry of hot and cold leg of the same SG. Regarding to complicated conditions for measurement in area of primary circuit, the overall uncertainty of monitored parameters evaluation is in the range of tens of percent of the measured value. Based on this assumption, the results of surface activities from two different methods are comparable.

IV. CONCLUSION

The radioactive corrosion products in Dukovany Nuclear Power Plant have been monitored during its whole operation. The data in TABLE III and Fig. 1 shows that the progress of surface radioactivity has had stable trend since year 2009 – about 3000 kBq/cm^2 with unit trend below value of 1000 kBq/cm^2 of summary surface activity. In conclusion, the radiation situation in Dukovany NPP is stable and on very low level. The chemistry control of cooling circuits significantly contributes to good radiation situation however the higher enrichment of new nuclear fuel might affect the RCP production.

TABLE III TRENDS OF SURFACE ACTIVITIES OF DUKOVANY NPP IN 2009–2018

	2009	**2010**	**2011**	**2012**	**2013**	**2014**	**2015**	**2016**	**2017**	**2018**
1. unit	963	880	1107	1670	930	813	808	808	510	552
2. unit	203	326	369	402	447	496	896	917	917	841
3. unit	2920	658	582	461	623	661	513	410	634	570
4. unit	481	809	538	878	602	456	585	585	620	538
All units	**4567**	**2673**	**2596**	**3411**	**2601**	**2426**	**2802**	**2720**	**2681**	**2501**

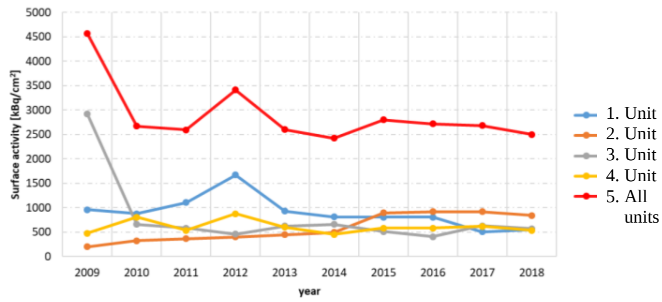

Fig. 1: Trends of surface activity of Dukovany NPP in 2009-2018 [6]

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TRACK 7: NUCLEAR FUEL CYCLE, WASTE MANAGEMENT AND DECOMMISSIONING

THE LONG-LIVED RADIOACTIVE WASTE AND USED FUEL MANAGEMENT IN SOUTH AFRICA: A LONG ROAD AHEAD

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THE REDUCTION OF URANIUM HEXAFLUORIDE WITH A ROOM TEMPERATURE IONIC LIQUID (1-METHYL-1-PROPYLPIPERIDINIUM BIS(TRIFLUOROMETHYLSULFONYL)IMIDE)

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FEASIBILITY STUDY OF PELLETIZED MATERIAL FOR THE BACKFILL OF DEEP GEOLOGICAL DISPOSAL FACILITIES FOR RADIOACTIVE WASTE

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BEHAVIOUR OF SPENT FUEL DURING STORAGE – ALMOST 40 YEARS OF COLLABORATIVE RESEARCH

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COUPLED THERMAL, HYDROLOGICAL AND CHEMICAL MODEL FOR TRANSPORT OF U(VI) IN ENGINEERED BARRIER SYSTEM AND HOST ROCK FOR A GENERIC NUCLEAR WASTE REPOSITORY CASE

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CHARACTERISATION AND VALIDATION OF THE WASTE CHEMISTRY FOR THE SYMO FACILITY

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EXPERIENCE AND EXPERTISE OF THE FUEL COMPANY TVEL IN DECOMMISSIONING OF NUCLEAR FACILITIES AND RADIOACTIVE WASTE MANAGEMENT

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THE NATIONAL RADIOACTIVE WASTE MANAGEMENT FACILITY PROJECT

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The Long-Lived Radioactive Waste and Used Fuel Management in South Africa:

A Long Road Ahead

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I. INTRODUCTION

Long-lived Waste (LLW) is defined as Radioactive waste containing radionuclides with half-lives greater than 31 years having sufficient radiotoxicity requiring long-term isolation from the biosphere. Used Nuclear Fuel (UNF) refers to the spent nuclear fuel removed from the three South African reactors in operation and classified as High Level Waste (HLW). Main generators of LLW in South Africa are the facilities involved with the uranium activities and its progeny. South Africa is well known for its gold resources and its nuclear history as the country was one of the founder member states of the International Atomic Energy Agency (IAEA). South Africa still operates the first nuclear reactor and the only Nuclear Power Plant (NPP) in Africa. The long mining history and operation of the nuclear reactors has led to the accumulation of LLW and UNF at the facility sites. Recognizing the importance of the safe management of UNF and radioactive waste, the South African Government has decided to implement remedial actions for LLW and UNF.

II. CONTEXT

A. Historical Background

The presence of radioactive material in the gold bearing ores of Witwatersrand basin was first noted in 1915 and in 1923 the material was identified as uraninite. In 1952, large scale uranium production became extensively popular in South Africa and is produced as a by-product of gold and copper mines [1]. The mining sector became the largest and most important part of the country's economy. For decades, the mining industry prospered without a radiological regulated framework and generated unregulated radioactive waste. In 1959, the Government approved the creation of a domestic nuclear industry and with cooperation of the US Atoms for Peace program, the SAFARI-1 research reactor went critical in 1965. The pool-type reactor initially fueled by Highly Enriched Uranium (HEU) has a maximum output of 20 MWt. SAFARI-1 is operated by the South Africa Nuclear Energy Corporation (Necsa) a state owned company which is used to carry out research fundamentally and applied science, industrial and medical isotope production. The reactor is located at the Pelindaba nuclear research centre near Pretoria. With the sanctions during the apartheid era, South Africa started to enrich the uranium locally to manufacture the fuel assemblies and to continue implementing the nuclear weapons programme. The uranium production plant contributed to the radioactive waste accumulated at Necsa.

In the mid-1970s the South African Government decided to build a NPP near from Cape Town. The two pressurised water reactors of Koeberg Nuclear Power Station (KNPS) were fully operational in 1985 and at present generating 4% of the country's electricity [2]. For the 40 year operational life of KNPS, the operator Eskom a state owned enterprise has implemented and maintained an adequate radioactive waste management programme which is a requirement from the Regulatory Body.

To manage the low and intermediate-level radioactive waste generated by KNPS and SAFARI-1 operations, an authorized disposal facility operated by Necsa was commissioned in 1986; Vaalputs situated in the arid region at approximately 500 km from Cape Town.

Figure 1. Geographical location of selected nuclear facilities

B. Regulatory History

Radiological regulation of the mining sector in South Africa started officially in 1990 through the establishment of the Council for Nuclear Safety. The mining sector was not familiar with radiation protection principles and showed a lack of appreciation for radiological risks. The operators were not used to be regulated and demonstrated a strong resistance to the implementation of regulations. Documented standards for regulating Naturally Occurring Radioactive Material (NORM) industries were also limited therefore the Government had to set up regulatory requirements.

The South African National Nuclear Regulator (NNR) was established in 1999 to provide for the protection of persons, property and the environment from the harmful effects arising from ionizing radiation produced by radioactive materials. The facilities and actions subjected to regulatory control by the NNR are diverse and include NORM activities such as mining and processing of radioactive ores. The NNR exercises its mandate over the NORM facilities through the NNR Act [3] and the Regulation R.388 [4] by issuing Certificates Of Registration (COR) or Certificates Of Exemption (COE) among other functions. To date, the NNR has issued more than 140 CORs to Holders which includes 69 Mining and Mineral Processing (Au, Cu, U, etc.) CORs and 19 CORs for scrap processors [5].

The NNR also regulates nuclear installations such as KNPS, SAFARI-1 and Vaalputs via Nuclear Installation Licenses. The regulation R.388 contains requirements which bound the Authorisation Holders in terms of exclusion and exemption criteria; radiation protection requirements, dose limits for workers and public, controls and limitations on operation and radioactive waste management.

III. Current Situation and Issues

A Mines

The gold discovery was a turning point in the history of South Africa. It changed the country from a principally agricultural society to grow into the world's largest producer of gold. The local gold sector started to decline in the early 21st century when the mines started to go deeper underground to reach the gold. This made the development of a mine to be very expensive, hazardous and challenging. In 2007, South Africa relegated to the 8^{th} position and became the 11th largest uranium producing country [6], which resulted to less revenue. Derelict and ownerless mines became the Government's problem over the years when owners disappeared, mining companies shirked responsibility towards environmental rehabilitation by leaving an area unrehabilitated prior to them being liquidated or leaving the country. "*Over the last century 600,000 tons of the radioactive uranium was inadvertently excavated through gold mining processes from the Witwatersrand basin. As for every 10g of gold excavated, 100g of uranium was also brought to the surface. Uranium is passed on to humans either through the inhalation of fine dust particles from these tailings and can be blown as far as 20km on a windy day, or when mine water seepage enters rivers*" explains North-West University Geographer Professor Frank Winde [7].

While international standard practice and local mining regulation prohibit locating residential areas closer than 500 m from a Tailings Storage Facility (TSF), it is estimated that 1.6 million people live in informal and formal settlements on, or directly next to, tailings in the Gauteng province [8].

Figure 2. Witwatersrand basin map and Dwellings proximity to TSFs

TSFs are some of the largest man-made structures, they are a lasting legacy of mining operations. Significant amount of residual process material comprised of the full uranium decay chain is released from mining and mineral processing facilities under waste rock. Another generator of large quantities of tailings that contain thorium is the mineral sands operations. Total accumulated waste, mainly from the gold mining industry, equals $2.23E^{+08}$tons [9] and scrap steel $5.45E^{+05}$tons [9].

(a) (b)

Figure 3. (a) Radioactive uranium waste lying exposed at Ryst Kuil
(b) Radioactive scrap metal from mining activities

B Pelindaba Site

The SAFARI-1 reactor will reach fifty-five years of operation in 2020, the long successful operation of the research reactor has resulted in an accumulation of UNF on the Pelindaba site. 957 UNF and 189 used control rods are temporary stored in the subsurface sealed stainless steel pipes in the Thabana Pipe Store after at least 2 year of pool cooling. Currently 216 UNF and 35 used control rods are stored in the pool [9].

Figure 4. Thabana Pipe Store at Pelindaba

The other high level waste is the lead test assemblies from the post reactor test fuel pin (0.25 m^3[9]).
The Pelindaba site hosts few storage facilities for LLW such as the decommissioned uranium enrichment facility which was transformed into a storage area for containerised solid radioactive waste from the historical Necsa nuclear fuel cycle programme. The total inventory for LLW classified as Low Level Intermediate Waste (LLIW-LL) is around 942 m^3[9].

Figure 5. Section of the storage facility at Pelindaba

Large volumes of these radioactive waste cannot be disposed at Vaalputs as they do not comply with the Waste Acceptance Criteria (WAC) of Vaalputs.

C Koeberg Nuclear Power Station

The current Government's focus is on the long term operation of KNPS [10]. Projects in support of the initiative includes the replacement of refueling water storage tanks and the 6 steam generators, these will contribute to the increase of radioactive waste on site. The revision to the 2016 Vaalputs WAC has introduced new requirements which impacted on the KNPS waste management programme. Delays in the shipment of radioactive waste has resulted in an accumulation of waste and increase in radiation levels at the on-site storage facility. The 35 years of KNPS operation have generated 112 UNF assemblies stored in 4 castors and 2177 UNF assemblies stored in the pools [8]. These high level waste do not comply with Vaalputs WAC and must be stored on site contributing to the risk to the plant of running out of wet and dry storage space.

Eskom has initiated a project to construct a new casks storage facility and have procured additional dual purpose casks. However, this project is a temporary solution as the country's position on UNF displayed in the 'Policy' [11] is clear: '*the storage of used fuel on a reactor site is not sustainable indefinitely*'. This statement is supported by the lessons learnt from the Fukushima accident as the UNF inventory must be limited on site [12].

IV. Remediation Strategies

The regulatory strategic plan for 2016-2020 [13] has established four actions for contaminated areas: a plan to develop remediation strategy; the incorporation of requirements for remediation, financial guarantees for decommissioning activities and to liaise with Department of Mineral Resource and Energy on the issue relating to national planning of remediation activities.

To enhance the control over NORM activities, the NNR has revisited its regulations to be in line with the IAEA Safety Standard GSR Part 3. The land remediation programme includes updated criteria for release of land remediated, other than exclusion and exemption criteria, the establishment of requirements/regulations for remediation activities and an authorisation process for remediation from a radiological point of view. To this effect, the Regulator had issued an interim guidance document [14]. The management option is to stabilize the mine tailings and waste rock by covering them with vegetation. This prevents loose soil from being blown or washed off by wind or water. Communities also, living in affected areas have to be relocated as the TSFs ultimately became disposal sites.

Figure 6. TSF rehabilitation programmes in Johannesburg

Long-lived waste containing residual uranium that couldn't be reprocessed are stored on the mine site. When the scrap metal is only contaminated on the surface, decontamination by high-pressure washing is performed otherwise the contaminated steel scrap is handed over to authorized facilities. Only a few facilities have authorization for processing contaminated scrap and only one facility is able to melt contaminated scrap steel. The waste management programmes of these facilities can be improved by upgrading and expanding their current capacity to process or melt contaminated scrap metal more effectively. Also provision should be made to construct more of these facilities to comfortably transport these scrap metals to the closet facility.

In accordance with the 'Policy', the Government is required to ensure that investigations are conducted within set timeframes to consider various options for the safe management of UNF and HLW in South Africa. To conduct the investigations, a National Radioactive Waste Disposal Institute (NRWDI) the 'Institute' was established and endorsed by the 2008 National Radioactive Waste Disposal Institute Act [15]. The Institute is an independent entity, reporting to the Minister of Mineral Resource and Energy, responsible for national radioactive waste disposal and amongst:

- Maintain a national radioactive waste database and publish a report on the inventory and location of all radioactive waste;
- Manage ownerless radioactive waste on behalf of the Government, including the development of radioactive waste management plans for such waste;
- Implement institutional control over closed repositories, including radiological monitoring and maintenance as appropriate.

These functions are currently performed within the scope of Low Level Waste Short Lived (LLW-SL) inventories and entail the Institute to become the new operator of Vaalputs.

Figure 7. Vaalputs disposal facility with LLIW-LL waste containers

One of the strategic objectives of the Institute is to develop and implement programmes for safe storage and disposal of UNF, HLW and long lived intermediate level waste (ILW-LL) on a national basis. In the absence of a deep geological repository for the disposal of UNF, the establishment of a Centralised Interim Storage Facility (CISF) for the long-term storage becomes a priority to comply with the ‘Policy’ [16]. At the present moment, the Institute cannot completely fulfil its mandate for various reasons, such as:

- Legal challenge. The responsibilities of the Institute may potentially overlap with the functions of others bodies.

- Regulatory challenge. The Institute must comply with the NNR requirements to be granted a nuclear installation license for Vaalputs.

- Institutional challenge. Transfer of function, budget, staff, assets and liabilities between Necsa the actual Vaalputs license holder and the Institute.

- Operating challenge. Existing infrastructure of Vaalputs had undergone ageing and has not kept pace with latest technology.

- Financial challenge. Currently, the Parliament provided restrained funding for the Institute’s establishment. The Radioactive Waste Management Fund (RWMF) based on ‘’polluter pays principle’’ must be finalized. The waste generators will contribute to the fund in equitable manner based on the classification and the volumes of the waste. The unit disposal cost (R/m^3) must still be established.

- Social challenge. The negative perception of radioactive waste by the public. Public and environmental Non-Governmental Organizations acceptance is the key to the success of the implementation of radioactive waste management strategies.

V CONCLUSION AND WAY FORWARD

South Africa hosts the largest amount of gold ore reserves in the world, but also most of the world’s deepest gold mines [17] making the extraction very expensive. The degradation of the economic climate motivates some of the mine owners to take shortcuts and to not fulfill their legal obligations. The lack of early regulatory oversight of the mining activities has resulted in uncontrolled production of LLW. The South African Government has to manage legacy and ownerless tailing sites occupied by human settlements. The land remediation programme includes protecting the sites and the issue of a COR to private companies for decontamination and rehabilitation once the waste management plan was approved by the NNR. This initiative is a success but has some limitations due to the cost and the number of people affected. The regulatory framework was revised and aligned with the best international practices and standards. A national Institute for disposal was established. However, the remediation actions planned by the South African Government face serious challenges due to various reasons. The investment in human resources and funds was scaled down after the suspension of the nuclear new built project. The priority is now on the disposal of all the HLW, UNF and LLW from future decommissioning activities. To fulfil this mission and also to be able to address the UNF and LLW stored at Necsa, the scope of the Institute must be expanded and a CISF must be established. The financial stability of the Institute will be achieved by its revenue from transfer payments received from government and other non-tax revenue received for providing waste disposal and related services to the waste generators, in particular Necsa and Eskom (KNPS). The Government needs to enhance the co-operation between the relevant State Organs with the establishment of agreements and improve awareness of the public through information forums and school outreach programmes. South Africa is on the right track to manage the UNF and LLW but still has a long road ahead.

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The Reduction of Uranium Hexafluoride with a Room Temperature Ionic Liquid (1-methyl-1-propylpiperidinium bis(trifluoromethylsulfonyl)imide)

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I. Introduction

Uranium hexafluoride (UF_6) is utilized in the fuel cycle for uranium enrichment based on the volatility, subliming at 56.4 ^oC [1]. Currently the United States Department of Energy has about 800,000 metric tons of depleted UF_6 waste in storage and there is a total of 1.2M metric tons worldwide. The primary method for conversion of UF_6 involves reacting gaseous with water vapor to produce UO_2F_2 and HF gas. The UO_2F_2 is further heated to 700 ^oC with water and H_2 to reduce it to U_3O_8 with additional HF offgas [2]. This study examines the viability of utilizing room temperature ionic liquids (RTILs) for the stabilization of UF_6 for transport, storage, and recovery.

RTILs are ionic solutions with negligible or low water content with melting points below 100 ^oC. RTILs have negligible vapor pressures, high thermal stabilities, are nonflammable and engineerable solubilities. RTILs can be advantageous in electrochemical studies based on the large electrochemical windows afforded by the lack of water and the high ionic conductivity. However metal solubility is typically low and their high viscosities inhibit diffusion of the target species [3 4].

In the present studies, 1-methyl-1-propylpiperidinium bis(trifluoromethylsulfonyl)imide ([MPPi][TFSI]) has been used. The spontaneous dissolution of depleted UF_6 into the [MPPi][TFSI] was examined. After dissolution, routes for uranium recovery as oxides and metal were explored.

II. Experimental

A. *Addition of UF_6 into [MPPi][TFSI]*

The [MPPi][TFSI] (99%) was purchased from Io-Li-Tec and opened only in a glove box under argon. The depleted UF_6, stored in Teflon tubes are chilled to liquid nitrogen temperatures before addition to the [MPPi][TFSI]. The temperature is sufficiently low that volatility is minimized. Quantities of [MPPi][TFSI] on the order of 5 to 20 mL of were placed in scintillation vials. The chilled UF_6 was added directly to the [MPPi][TFSI] in the vial. The solution was then stirred for 24 hours with a stir bar and plate to accelerate the dissolution.

B. *Electrochemical Methods Used*

Electrochemical depositions were performed using a CHI 760C potentiostat. A three electrode cell was utilized with a gold foil working electrode, platinum foil counter electrode, and pseudo-reference electrode made with a silver wire and $Ag(NO_3)$ in [MPPi][TFSI]. The reference electrode offset was measured using the redox couple of ferrocene. Typical cell volumes of 5 mL of a 0.1 M solution of UF_6 in [MPPi][TFSI] were utilized. The working electrode was held at a potential of -3.5 V to achieve deposition.

C. *Instrumentation*

UV-Vis data was collected using a Cary 6000 with Cary Win Software. X-ray absorption fine structure (XAFS) measurements were completed at the Advanced Photon Source at BESSRC-CAT 12 BM station at Argonne National Lab. XAFS spectra were recorded using the U-L3edge (17.166 eV) in transmission mode. Data was background subtracted and analyzed using the Athena and WINXAS software. Thermal gravimetric analysis (TGA) was completed under argon using a Netzsch Jupiter STA 449 F1 coupled TGA-DSC. Powder x-ray diffraction (PXRD) was collected using a Bruker D8 Advance. Scanning electron microscopy (SEM) images were taken with a Jeol JSM-5610.

III. Results and Discussion

A. *Dissolution mechanism of UF_6 in [MPPi][TFSI]*

The dissolution of UF_6 into the [MPPi][TFSI] can be visually observed after addition to the RTIL. The white solid turns yellow as soon as it is added to the [MPPi][TFSI], and then rapidly to aquamarine. Over time the solution changes to green. While stirring is used to assists in the dissolution of low concentration solutions, higher concentrations over 0.5 M, dissolve without. The process is exothermic and at higher concentrations the heat generated can be felt. The exothermic

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nature of the dissolution assists in accelerating the dissolution of the high concentration solutions.

UV-Vis spectra were collected over time for an 80 mM solution to evaluate color changes. The spectra for the solution after 4 hours, 24 hours and 8 days are presented in Figure 1. Although there are minor changes in peak intensity of several peaks over the course of the week, the changes are minimal. The data suggests that the dissolution product is stable in the IL under the time range studied. Measurement below 500 nm is difficult because [MPPi][TFSI] strongly absorbs in the UV range. However, the shoulder of the spectra in the UV range shifts over time suggesting that there are interactions that between the [MPPi][TFSI] and UF_6 over time. The peak at 640 nm increases over time whiles the peaks at 700 and 760 nm decrease. Lastly, there are significant decreases in the peak at 1370 nm.

XAFS data of a 10 day old sample shows 5.8 ± 1.2 F atoms at 2.05 Å from the uranium. The 2.05 Å bond distance suggests that the complex is possibly UF_6^- with the uranium in the +5 oxidation state. In 1967, a pentavalent UF_6 complex, $CsUF_6$, was reported with a U-F bond length of 2.057 Å [5]. In 1979 the quadravalent δ-Na_2UF_6 crystal was reported with U-F bond lengths ranging from 2.23 to 2.42 Å [6]. However additional studies are planned to determine if UF_6^{2-} is also present in the IL.

Based on the XAFS a proposed mechanism for dissolution involves electron transfer from the [TFSI] anion to reduce UF_6 from U(VI) to U(V). The [TFSI•] free radical is stable and has been reported in literature previously [7]. It is possible over time the [MPPi] cation coordinates with the dissolved UF_6^- anion to make a [MPPi][UF_6] complex. This interaction might influence the small change in spectroscopic properties observed in the UV-Vis.

B. Precipatation of [MPPi]2UF6

In high concentration solutions of UF_6 in [MPPi][TFSI] (>0.5 M) a precipitate forms over the course of a couple weeks. It is suspected that a second electron transfer occurs to reduce to U(IV). As a second [MPPi] cation coordinates to the U, the bulky $[MPPi]_2[UF_6]$ complex drops out of solution. The precipitate can be filtered and cleaned of the [MPPi][TFSI] with acetone. The precipitate appears to be air stable over extended periods of time at room temperature.

TGA data of the precipitate was obtained for three samples, Figure 2. The initial mass loss at 400 ^oC was 42 to 44% for the samples. This corresponds to the loss of the 2 [MPPi] cations equivalent to 44.7% mass loss. The second mass loss at 1100 ^oC is 56 to 59% for the samples. The loss correlates with 6 fluorides and the addition of O_2 and has a mass equivalent to 57.6%. The data suggests that higher concentrations of UF_6 dissolved in RTIL produces U(IV) which can then complex with two cations to form the precipitate. PXRD was completed on the final powder to confirm the production of UO_2 and the mass of U in the sample.

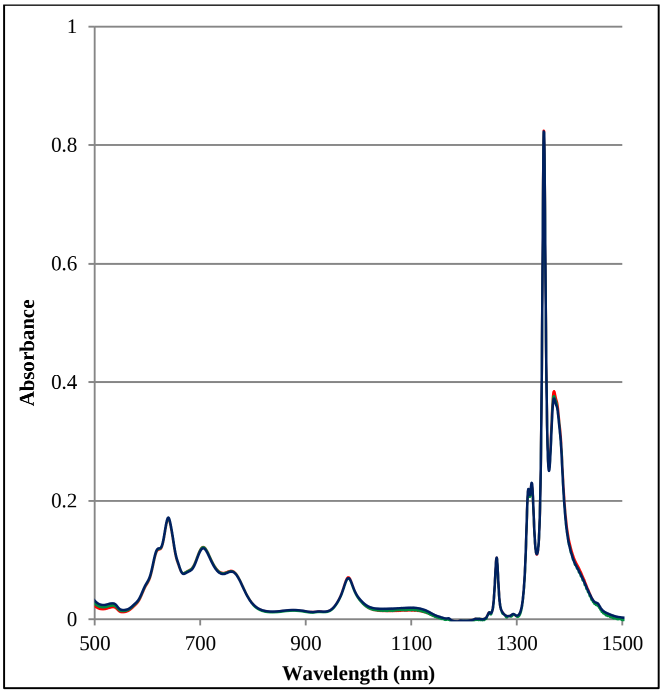

Figure 1. UV-Vis of 80mM UF_6 solution at 4 hours in red, 24 hours in green, and 8 days in blue.

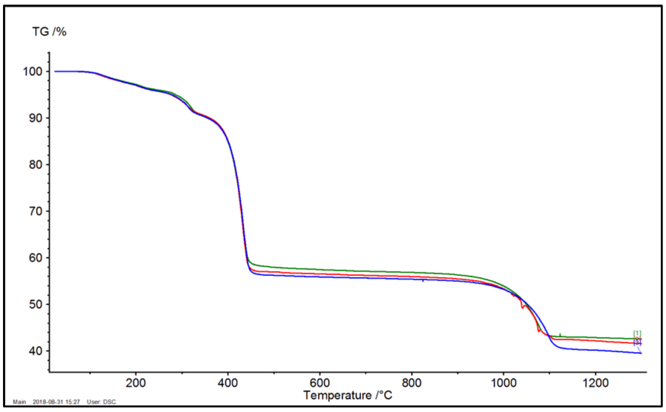

Figure 2. Thermal gravimetric analysis of precipitates from high concentration UF_6 solutions.

C. Electrochemical reduction of Uranium

The electrochemical reduction of U species from the RTIL solution containing 0.1 M UF_6 was examined. The goal was to convert the soluble uranium species to metal. The gold electrode was held at -3.5 V for 40 hours to fully reduce the uranium species. Dark deposits on the gold foil electrode were obtained that were dendritic in nature and sluffed off the electrode into the [MPPi][TFSI] when the electrode was removed from the solution. Although there were no visible deposits remaining after removal from the RTIL, SEM analysis confirmed small deposits on the electrode surface (Figure 3). Energy dispersive x-ray spectroscopy on the electrode showed the deposits to be uranium. There was also some fluorine and sulfur present on the electrode, which may be residual from the [MPPi][TFSI]. However, oxygen was not observed which indicates the deposits are not oxides.

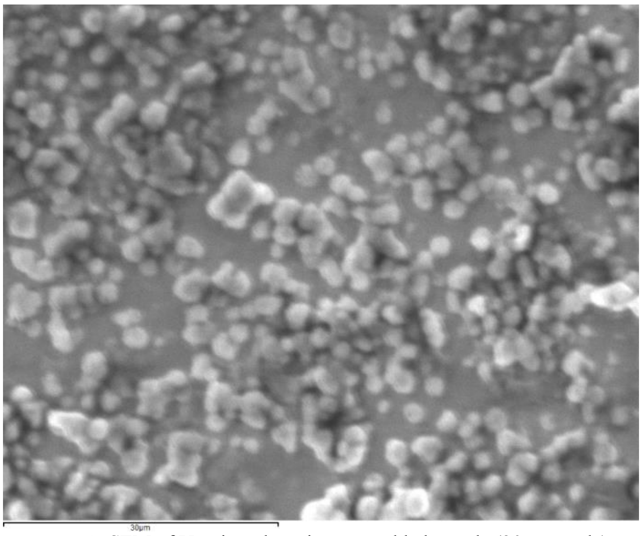

Figure 3. SEM of Uranium deposits on a gold electrode (30 µm scale).

IV. CONCLUSION

The addition of UF_6 to [MPPi][TFSI] has been shown to produce a stable solutions of complexed UF_6 suitable for storage and transportation. The direct addition is a simple method to neutralize UF_6 and has the potential for larger scale applications. Uranium recovery is feasible in the form of a precipitate that can be converted through heating in atmosphere to UO_2. In further studies, heating of the precipitate under a reducing atmosphere with H_2 will be explored to produce uranium metal. There is also potential to electrochemically reduce the uranium from the RTIL solution to metal on much smaller scales. Optimization of the electrochemical processes for deposition may allow for higher recovery rates at the electrode interface.

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Feasibility study of pelletized material for the backfill of deep geological disposal facilities for radioactive waste

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I. INTRODUCTION

In France, High-Level (HLW) and Intermediate-Level, Long-Lived (ILW-LL) radioactive waste will be placed in a deep geological disposal (Cigéo) located at about 500 m deep in a clay formation (Callovo-Oxfordian). Confinement of the radionuclides in the long term relies mainly on the transport properties of the clay formation but also on the progressive closure of the galleries with sealing plugs installation as well as by backfilling the structure's remaining openings [1]. A high density upon the sealing/backfill materials emplacement is of high importance, since decreased initial voids will improve the hydro-mechanical (HM) response upon hydration from the saturated surrounding rock.

A possible solution suggests the backfill material to be composed of excavated crushed COx argillite and swelling-clay additive (MX80 sodium bentonite), formed into pellets (i.e. highly compacted granules of quasi-spherical shape) in various sizes. This suggestion is advantageous in various aspects. It exhibits better behavior upon implementation, since the dusty conditions produced by an in situ compaction on a deep excavation project can be avoided by the use of pelletized material. On the other hand, a mixture of COx and MX80 inside the pellet benefits the homogeneity and avoids a possible risk of argillite behaving as liquid at saturation. The pelletized mixture will be placed at almost dry conditions and will behave as a granular medium before hydration. However, as water absorption occurs, homogenization starts taking place [2, 3]. The emplaced density depends on several factors such as pellet's density, grain size distribution (GSD) of the pelletized mixture, surface characteristics of the different sized pellets as well as the implementation protocol.

The study focuses on the material's HM behavior, in order to determine (i) the optimum grain size distribution (GSD) that facilitates high implementation densities and (ii) the material's composition (%COx, %MX80) which meets up the design requirements in terms of swelling potential and permeability. The mixture of COx and MX80 will be pelletized at high dry density (around 2.0 g/cm^3). In the present work, the optimum GSD is studied by conducting numerical simulations (DEM) on samples with various granulometries. Afterwards, fabricated pellets corresponding to the chosen GSD are used in order to evaluate experimentally the compaction state of the pelletized mixture. Finally, swelling under constant vertical stress tests investigate the swelling capacity of mixtures composed of COx argillite and MX80 bentonite hydrated by synthetic site water.

II. OPTIMISATON OF CHOSEN GRAIN SIZE DISTRIBUTION

A. Used model

The study concerning the optimum GSD focuses on the compactness of a pelletized mixture, by simulating in 3D conditions the deposit of a sample containing spheres inside a cubic cell ($l_x=l_y=l_z$), under the effect of gravity. The method primarily studies the granulometric effect on porosity, by examining the impact of the volumetric proportion of each sphere on the resulted density state. In this concept, the Discrete Element Method (DEM) was chosen. Pellets are considered as rigid spheres. Coulomb's law of friction was selected for the case of dry contact with non-cohesive bonds, in order to define the behavior between the spherical particles. A non-cohesive model was chosen due to the coarse size of the granular material [4]. The cell dimensions take into account the maximum diameter of the generated sample ($l_x=l_y=l_z=10d_{max}$). The open-source software LMGC^{90} is chosen to run the simulations. The software implements the DEM method using the theory of Non-Smooth Contact Dynamics (NSCD), which obtains the overall behavior of the medium by considering the dynamics of each element, taking into account the interactions between bodies [5].

The spherical particles are randomly generated according to an imposed GSD (diameters d_i, volumetric proportions $\%d_i$) inside the five planes of a cubic cell. The initial height of the specimen is 1.25 times larger than the box's height (l_z) to generate a loose initial packing state (Fig. 1a). The sixth plane is in contact with the highest sphere; it has no mass and allows determining of the sample's volume. The sample is left to fall from a given height, h under the gravity effect and the particles are rearranged, until the kinetic energy is turned into zero. The final density is calculated taking into account the solid and total volume the end of the deposit (Method I) (Fig. 1b). In order to eliminate the edge effects and decrease a possible influence of the cell's scale on the packing state, a fictional spherical wall is considered and the final density is re-calculated by taking into account only the spheres intersecting the wall with centers located in its interior (Method II) (Fig. 1c). It is an attempt to simulate better the underground gallery where the backfill will be emplaced. This method proved to be close to an analytical method proposed by Marshall and Dhir [6].

Different sized spheres are selected to be tested (d_{max}, d_{med}, d_{min}). The sizes are compatible with the pelletizing process and the laboratory scale where will be later studied. Various volumetric percentages are tested. A non-zero but relatively low friction coefficient describes the interactions between the particles (μ=tan5°=0.1). The friction between particles and cell are neglected. As default, the specimens are deposited from zero height, expecting to result in a lower packing density. In this way, computation time is eliminated and uncertainties caused by the low friction parameters could be balanced. The density of the spherical particles is always considered equal to 2 g/cm^3.

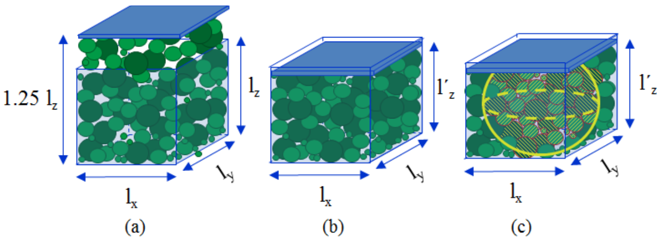

Figure 1. DEM results: presentation of the(a) sample generation, (b) end of the deposit and (c) fictional spherical wall. The achieved denstiy was calculated according to Method I (b) and Method II (c).

B. *Achieved density of trimodal samples*

Several ternary mixtures were tested. Table I summarizes the geometrical characteristics of the investigated diameters. In the study, size ratio, α_i is defined as the ratio of the smaller diameters to the maximum and expresses the gradation of the granular material. Lower size ratio results in larger range of particle sizes. In total, 17 volumetric proportions were tested for each sample.

The results of the conducted simulations are presented on the graphs of Fig. 2. The obtained dry density as determined according to method II is illustrated for the proportion for maximum and minimum tested diameter. An effect of the GSD on the final density is observed. Generally, it can be concluded that higher densities are resulted when the proportion of the larger particles is 60-70% and the proportion of the smaller particles is 10-30% for each sample, while the proportion of the medium sphere was the least significant. Standish and Borger [7] also reported that low porosities are achieved when the proportion of small particles is intermediate.

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TABLE I. INVESTIGATED DIAMETERS OF TRIMODAL SAMPLES

Diameters (mm) d_{max} / d_{med} / d_{min}	***Size ratio**** α_{max} / α_{med} / α_{min}
12 / 7 / 4	1.0 / 0.58 / 0.33
16 / 10 / 4	1.0 / 0.63 / 0.25
20 / 10 / 5	1.0 / 0.50 / 0.25
12 / 4 / 3	1.0 / 0.33 / 0.25

* Size ratio: a_i=d_i/d_{max}

On Table II, the densest state for each trimodal sample is given. The global dry density obtained according to Method I is also illustrated. Except for sample of 12/7/4 mm, the rest trimodal samples exhibit higher densities with low variations (for both Methods). This can be justified as a result of the common decreased size ratio of the minimum diameter α_{min}. Similar behavior regarding the size ratio has been observed on bimodal mixtures by Standish and Yu [8]. In these cases, the smaller particles can easier fill the voids creating by the two larger diameters. On the other hand, the variation between the two methods clearly points out the effect of the edge effects which is related to the scale of the test cell. As optimum GSD is considered the trimodal sample containing spheres of 16 mm, 10 mm and 4 mm. It exhibits one of the highest densities and the specific sizes are more compatible with the capacity of pellet machine used for their fabrication.

III. PELLET FABRICATION

The pellets are compacted by means of a press machine. An adapted tablet press Frogerais type 1B (Fig. 3) is used to fabricate the chosen GSD. The raw material will be inserted into molds of the desired diameters and will be compacted by means of pistons, while punch holes will give its final pellet shape. The diameters of 16 mm, 10 mm and 4 mm are designed for the fabrication of the chosen GSD. Firstly, mixtures containing 30% and 40% of MX80 were compacted at the initial granular form (d_{max}=2 mm) (Fig. 4a, 4b) into pellets (Fig. 4c). The achieved dry density was measured by immersing a pellet covered with paraffin inside water. A dry density of at least 1.90 g/cm^3 could be successfully obtained.

A. *Achieved density of bimodal peletized sample*

The density state of a bimodal sample containing pellets of 16 mm and 10 mm was tested. For this purpose, a transparent cubic cell, composed of plexiglas planes with dimension l_x=l_y=160 mm and l_z=180 mm was fabricated. The dimensions were chosen to be in agreement with the cell used on the numerical simulations. A slightly higher height was chosen for a larger volume of specimen. Several proportions (50/50, 60/40, 70/30, 80/20 in mass) of the two diameters were mixed and the achieved dry density was measured. Totally, 5 tests were carried out. The results were compared with results of the numerical simulations for the same diameters, proportions and test numbers.

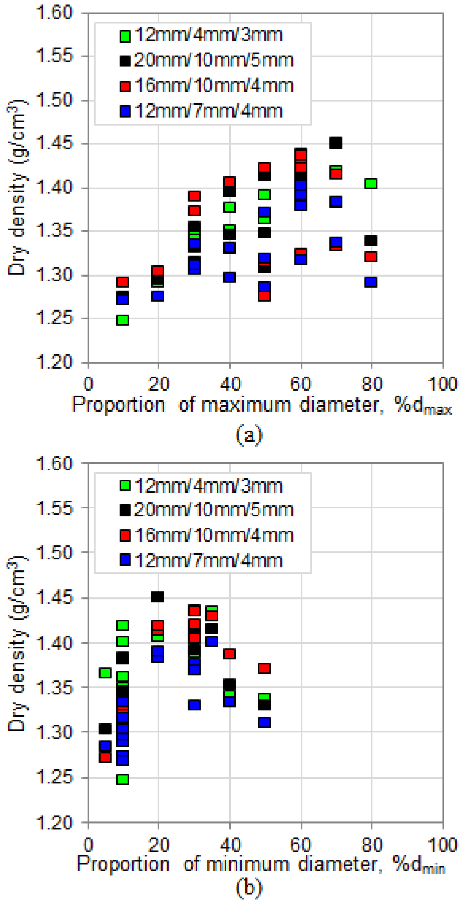

Figure 2. Variation of dry density of trimodal samples in relation to (a) proportion of maximum and (b) proportion of minimum diameter.

TABLE II. DENSEST STATE FOR EACH TESTED TRIMODAL SAMPLES.

Diameters (mm) d_{max} / d_{med} / d_{min}	*Size ratio* α_{max} / α_{med} / α_{min}	*Dry density Method II (g/cm^3)*	*Dry density Method I (g/cm^3)*
12 / 7 / 4	1.0 / 0.58 / 0.33	1.40	1.32
16 / 10 / 4	1.0 / 0.63 / 0.25	1.44	1.38
20 / 10 / 5	1.0 / 0.50 / 0.25	1.45	1.36
12 / 4 / 3	1.0 / 0.33 / 0.25	1.45	1.38

Fig. 5 illustrates the obtained average dry density of the conducted tests, as well as their deviation. A slight variation in density is observed for every case (experimental and numerical), which remains less 1.2%. An important difference between the two methods of numerical calculations and the experimentally obtained results is detected. The experimental results are close to those obtained by the global density at the end of the deposit (Method I) (absolute error: 0.28-1.02%). On the other hand, the method considering the edge effects (Method II) gives higher deviation (absolute error: 7.31-10.23%). The observation suggests that the scale of the test cell might significantly affect the compaction state, since the density increases 8-9% with the elimination of the edge effect.

Figure 3. Tablet press Frogerais type 1B for pellets fabrication.

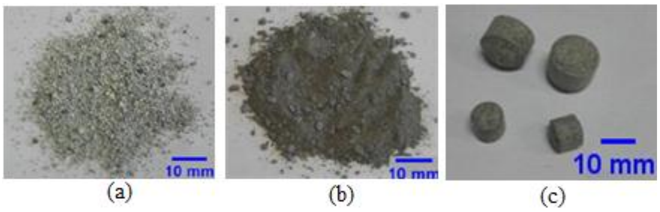

Figure 4. Images of (a) MX80 bentonite, (b) COx argilite, both at the initial form and (c) fabricated 16 mm and 10 mm pellets.

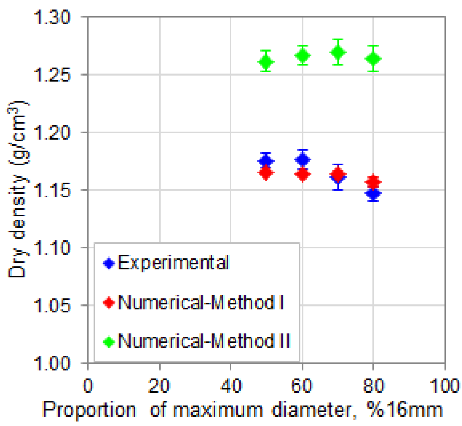

Figure 5. Variation of numerical and experimental dry density of bimodal sample containing 16 mm and 10 mm in relation to proportion of 16 mm.

IV. SWELLING CAPACITY OF COX/MX80 MIXTURES

In the disposal facilities, the backfill installation will be followed by its hydration from the surrounding COx claystone. In the beginning, the material will absorb water and swell under almost free volume conditions, until the initial voids are filled. The COx/MX80 mixtures (≤40%MX80) capacity of recovering initial void was studied by performing swelling under constant vertical stress tests. The two materials were compacted at the initial granular form (Fig. 4a, 4b) until a dry density of 1.45 g/cm^3 was achieved, corresponding to the optimum GSD density (Method II). A low vertical stress of 6 kPa was applied and specimens (d=40 mm, h≈20 mm) were directly hydrated with synthetic site water (Table III) under a low injection pressure (P_{inj}=4 kPa). The volumetric deformations were recorded by means of a dial gauge. The deformation developed upon time is illustrated at Fig. 6. As expected, a clear impact of the bentonite content on the swelling response is observed. The swelling is developed hyperbolically with time. For every tested mixture, a recovery of initial voids ranging from 3.8% to 18.3% is expected for initial dry density of 1.45 g/cm^3.

TABLE III. CHEMICAL COMPOSITION OF SYNTHETIC SITE WATER.

	Chemical components						
	NaCl	***$NaHCO_3$***	***KCl***	***$CaSO_4$ $2H_20$***	***$MgSO_4$ $7H_20$***	***$CaCl_2$ $2H_20$***	***Na_2SO_4***
g/L	1.95	0.13	0.04	0.63	1.02	0.08	0.70

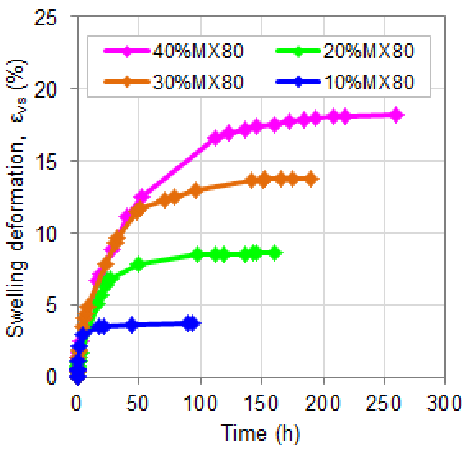

Figure 6. Swelling deformation under 6 kPa for mixture of COx argilite with 10%, 20%, 30% and 40% of MX80 bentonite compacted at initial density of 1.45 g/cm^3 using site water.

V. CONCLUSIONS

The implementation feasibility of a pelletized material used as backfill for the radioactive waste disposal was studied. The obtained results can be summarized as following:

-The optimum GSD was investigated by studying the impact of granulometry on the materials compactness. The density is determined using two methods: global data obtained at the end of the sample deposit in the cubic cell (I) and by a method eliminating possible edge effects closer to the large scale of the underground galleries (II).

-The densest state on trimodal samples is achieved when the following proportions criteria are fulfilled: $\%d_{max}>\%d_{min}>\%d_{med}$. Decrease in size ratio α_i results in higher compaction. However, parameter α_{min} is more influential, because the smaller particles can better penetrate inside the voids. When the same α_{min} is imposed, low variations in the achieved density are observed. The two methods (I, II) reveal an influence of the cell's scale on the compaction state. According to density performance and pelletizing process capacity, the spheres of 16 mm, 10 mm, 4 mm is chosen in proportions of 60%, 10% and 30% respectively.

-An adapted tablet press fabricated pellets of 16 mm and 10 mm. Experiments on the achieved density of the bimodal sample showed good agreement with the numerical density of Method I (absolute error 1%). Test scale probably affects the resulted compaction state and measurements on larger scale should be carried out in the future.

- The capacity of void recovery was evaluated on mixtures of COx/MX80 (≤40%MX80) at the initial granular non-pelletized form. Their compaction state corresponded to the one obtained for the optimum GSD (1.45 g/cm^3). A recovery of initial voids ranging from 3.8% to 18.3% is observed. As expected, swelling deformation clearly increases as bentonite content increases.

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Behaviour of spent fuel during storage – almost 40 years of collaborative research

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I. INTRODUCTION

Since 1981, the International Atomic Energy Agency (IAEA) has organized Coordinated Research Projects (CRPs) to support the long term storage of spent nuclear fuel from power reactors.

These CRPs gather research and development (R&D) results from the participating countries relating to the technical issues associated with storing spent fuel from all types of commercial power reactors through the IAEA project TEChnical DOCuments (TECDOCs). As well as documenting the changing approach to managing the back end of the fuel cycle and the resolution of past technical issues, the projects have facilitated the transfer of knowledge from one generation of experts to the next.

This paper covers the key findings of a series of CRPs completed under the titles of 'BEFAST' (BEhaviour of spent Fuel Assemblies during extended STorage) [1–3] and 'SPAR' (Spent fuel Performance Assessment and Research) [4–6]. The SPAR-III & IV phases ran concurrently with another CRP entitled 'Demonstrating performance of spent fuel and related storage components during very long term storage' (DEMO). (2013-2016) that focused on small to large scale testing to support long term dry storage of LWR fuel [7].

The future areas of focus for IAEA collaborative efforts in spent fuel storage are also outlined.

II. THE ORIGINS OF BEFAST AND SPAR PROJECTS

In 1979, the IAEA initiated a survey of water reactor spent fuel storage experience. The evidence obtained provided a positive basis for regarding wet storage of spent fuel from Light Water Reactors (LWRs) and Heavy Water Reactors (HWRs) as a "proven, viable technology" [8].

The report from the International Nuclear Fuel Cycle Evaluation (INFCE) Working Group that reported its findings at a similar time were in agreement and recommended "observation and investigation should be continued to evaluate the behaviour of high burnup spent fuel assemblies during prolonged storage periods and to confirm the present positive experience" [9].

III. BEFAST (1981-1996)

BEFAST-I was launched in 1981 involving 11 institutes from 10 Member States (MSs) to fulfil this INFCE WG recommendation [1]. The original aim of the project was to provide a significant database on the cladding integrity of spent fuel after extended storage periods. Five areas of interest were identified:

- Survey existing spent fuel storage experience;
- Destructive examination of spent fuel both before and after storage;
- Investigation of potential cladding degradation mechanisms;
- Evaluation of non-destructive techniques for storage;
- Investigation of the behaviour of spent fuel pool equipment.

Following completion of the project in 1986, further areas of research for the improvement of the technology and reduction of the costs of storage facilities were recommended. These recommendations were followed up in the subsequent BEFAST-II (1986–1991) and BEFAST-III (1991–1996) projects.

BEFAST-II brought experts from four new MSs and expanded the range of fuels to those from gas-cooled reactors (GCRs) [2]. This second phase was organized to cover three major topics:

- Long term behaviour of spent fuel;
- Surveillance of spent fuel;
- Spent fuel storage facilities and operation.

As the project reached a third phase in BEFAST-III, the R&D focus was shifting from wet storage to the performance of fuels stored in dry technologies [3]. Other general trends were observed, including the steady increase of fuel burnup and increasing storage durations. The individual projects at this time started to focus on the operation of facilities, the implementation of new design facilities and technological improvements. Even though many of the R&D questions relating to the basic material science had been answered by this point, extrapolation of this data for the >50 years storage durations envisaged still required confirmation.

IV. SPAR (1997-2020)

As fuel burnups increased, so did the fission gas content and fission product inventories, cladding strains and amount of cladding oxidation and hydriding. To address the impacts of these parameters on long term storage and to determine their consequences on post-storage operations, such as transportation, SPAR was initiated. Unlike many of the CRPs organized by the IAEA, SPAR was purposefully designed to have a very broad scope, covering all types of power reactor fuel under both wet and dry storage conditions. It also included research relating to the storage systems themselves and how they performed under such conditions. The SPAR projects also covered topical issues in spent fuel storage as they arose.

The four SPAR phases have involved between 10–12 MSs and one International Organization, with 11–15 institutes participating within each phase.

- SPAR-I (1997-2002) [4];
- SPAR-II (2004-2008) [5];
- SPAR-III (2009-2014) [6];
- SPAR-IV (2016-2020)

V. DISSEMINATION OF RESEARCH

With the completion of each CRP, an IAEA TECDOC was released documenting the findings from each of the participating institutes. These TECDOCs are still available from IAEA but over time many of the questions raised in earlier phases have since been resolved by subsequent research, rendering some of the original conclusions obsolete in isolation. To ensure continued dissemination of relevant information, the findings from BEFAST I–III and SPAR-I–III were reviewed and consolidated into a single TECDOC, providing a single referenceable source for the work completed under the programme between 1981–2014 [10]. The TECDOC from the fourth phase of SPAR is currently in production and should be available late 2020.

VI. OBSERVED TRENDS IN SPENT FUEL STORAGE

Looking back over the almost 40 years of contributed research, a number of trends in spent fuel storage are evident.

A. *Spent fuel storage durations are increasing*

Over the past 40 years, the planned durations for spent fuel storage have been gradually increasing. This in part has been due to delays in implementing the disposition steps (disposal and/or recycling) of spent fuel but increasing knowledge and experience in the behaviour of spent fuel over longer storage durations has given confidence to increased storage durations.

B. *Development of commercial spent fuel dry storage*

When BEFAST-I was initiated, with the exception of the Wylfa Nuclear Power Plant (NPP) in the United Kingdom, the only commercial spent fuel storage systems in operation were wet. Demonstration trials for LWR spent fuel dry storage began in the late 1970s, but it was not until the mid-1980s that systems were deployed. Now, most new away from reactor (AFR) systems deployed utilize dry technology.

One example of this is spent fuel storage trends in United States of America. At the inception of BEFAST, all fuel was stored wet in pools. By 2016, over one-third of the national inventory was stored in dry systems and this proportion continues to grow.

C. *Increasing burnup of spent fuel*

To maximize the potential of fuel, burnups have been increasing over time. The maximum burnups achieved vary by country depending on regulatory limits, and reactor operational licenses, fuel enrichment and operational cycle times. As an example, in the USA during the early 1980s, typical burnups achieved for Pressurized Water Reactor (PWR) spent fuel were in the region of 20–40 GWd/t. Today, average burnups are in the region of 40–55 GWd/t.

New fuels, such as Advanced Technology Fuels (ATFs) are anticipated to reach burnups of up to 75 GWd/t using fuel enriched up to 7.5% ^{235}U [11].

VII. RESEARCH FINDINGS

Unless otherwise stated, the findings here are given in the context of zirconium clad LWR fuels.

A. *Wet storage*

Over the seven project phases, three potential degradation mechanisms that may affect cladding integrity during wet storage have been investigated.

1) *Uniform aqueous corrosion*

In spent fuel pool water maintained at 30–40°C, the corrosion rates in water are around 10^{10} lower compared to the corrosion rates experienced under operational reactor conditions. As a result, the amount of corrosion experienced by zirconium alloys during one full power day is equivalent to the corrosion experienced over several million years of wet storage.

Other materials, such as stainless steel and Inconel, experience higher corrosion rates in water than zirconium alloys, but still demonstrate excellent durability in wet storage conditions, with corrosion rates less than 1 μm/year.

2) *Localized corrosion*

Here are several types of localized corrosion that can be experienced during wet storage, namely galvanic, pitting or microbial induced corrosion (MIC). Research has found galvanic corrosion is unlikely for zirconium clad fuel as they are found either to be in contact with aluminium, which also is readily passivated by oxidation, or with electrically similar stainless steel.

Under normal pool conditions, zirconium alloys have been found to not be susceptible to pitting type corrosion.

No MIC of zirconium alloys (or pool storage equipment) has been observed. There is however evidence for MIC in cooling structures.

3) Hydriding

When cooled to pool temperatures, the hydrogen present in zirconium alloy precipitates in the form of hydride platelets. Due to the heat transfer characteristics of water, hydrogen distribution leading to the formation of pits by Ostwald ripening can be ruled out during wet storage.

B. Dry storage

Six potential degradation mechanisms that can affect cladding integrity during dry storage have been investigation, and in some instances ruled out under normal storage conditions.

1) Air oxidation

Significant quantities of air in the presence of fuel in normal inert dry storage conditions can be ruled out by appropriate drying and inerting operations. Under these conditions, oxidation is not an active degradation mechanism.

For instances like a seal failure, there is a possibility for it to be an issue for off-normal or accident conditions. For intact fuel, any loss of cladding thickness would be limited by the low rates of zirconium oxidation by air, even in the case of prolonged exposure. It is really only an issue where through-wall defects are present in the cladding at temperatures exceeding 250°C. Above these temperatures, the conversion of UO_2 to U_3O_8 is possible, leading to a volume expansion and high cladding stress levels that may lead to gross cladding rupture.

2) Thermal creep

Thermal creep is the deformation of material at high temperature and under strain over time. It has attracted a lot of research attention and there is wide regulatory guidance in relation to it. The interaction between the fuel pellet and the cladding itself has proven to be an important behavioural characteristic, leading to a requirement to carefully interpolate results gathered from trials using empty cladding tubes or cladding samples alone.

Creep deformation is a slow process, and as the fuel cools over time, the internal rod pressure will decrease and further slow the progression of creep. Under normal conditions of storage (max temperature <400°C), creep should not result in gross cladding rupture.

3) Stress corrosion cracking (SCC)

SCC requires the presence of an SCC agent such as iodine, a specific high temperature range and cladding stress conditions for crack propagation to occur. Under normal storage conditions, cladding failure via this mechanism is not expected to occur.

4) Delayed hydride cracking (DHC)

DHC is a very specific mechanistic process, including dilation of the zirconium alloy crystal lattice structure by triaxial stressing. The cladding tubes tend to have insufficient thickness to generate the triaxial stresses required, so this form of cladding failure is not expected in zirconium alloys.

5) Hydride reorientation

During the fabrication of zirconium cladding tubes and reactor operations, circumferential hydride platelets can form in the cladding. If the spent fuel is transferred into dry storage, a rise in cladding temperature can cause dissolution of these hydrides up to the hydrogen solubility limit. As the fuel cools during dry storage, the hydrides reprecipitate. But due to the influence of cladding stresses, precipitation can be in a radial direction. The mixed hydride structure (both radial and circumferential) can significantly affect the mechanical properties of the cladding and its failure limits, especially at temperatures below 200°C. This can have implications for operations and handling post-storage.

Due to the potentially adverse effect of radial hydride formation on the ability to maintain cladding mechanical integrity under off-normal conditions, a lot of research effort is continuing in this field to develop a deeper understanding of any potential consequences for future spent fuel handling.

VIII. KNOWLEDGE TRANSFER

Over the course of the BEFAST and SPAR programmes, 21 countries and one international organization have provided experts to participate in the exchange of research and operational experience relating to spent fuel from all commercial power reactors. This type of exchange is particularly beneficial where a particular institute does not have the resources required to conduct particular work themselves but can share in the results of others who do have. It also enables participants to identify where a collaborative approach to a single problem could yield results more efficiently.

In addition, given the long timespan of the programmes, there has been a gradual turnover of experts as older members have retired and new members have joined the team. This has ensured a means to pass on detailed technical knowledge between the original generation of experts and the next.

IX. THE FUTURE

Despite the conclusion of the SPAR programme, continued collaboration of research and sharing of experiences relating to spent fuel storage remains important. As of 2021, the IAEA will be launching a new CRP to capture the continuing R&D pertaining to SPAR and DEMO, covering the performance of spent fuel and storage systems (wet and dry) during long term storage and transport, including experiences from demonstration programmes.

X. CONCLUSIONS

Through the BEFAST and SPAR CRPs, the IAEA has been building a technical knowledge base on the performance of spent fuel and materials in both wet and dry storage conditions for almost 40 years. The programmes have benefitted from the contributions of 21 MSs and one International Organization.

Given the trend for increasing durations of spent fuel storage, changes to the operation of reactors to increase efficiency and the introduction of advanced fuels, it is still important to confirm the viability of long term spent fuel storage and its subsequent transportability through efforts such as these CRPs. The new CRP commencing in 2021 aims to achieve this by bringing together the community to share data on fuel and system performance and by identifying future topics that could benefit from focused research.

ACKNOWLEDGMENT

The authors wish to thank all of the contributors to the BEFAST and SPAR projects, including previous IAEA scientific secretaries. In addition, special mention is given to Ferenc Takats, who has been involved in the projects since BEFAST-II as a participant, IAEA scientific secretary and as Chair of SPAR I-IV.

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Nuclear Material Retrieval Best Practices and Lessons Learned from the Legacy ^{252}Cf Sealed Source Closeout Project

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I. INTRODUCTION

The United States Department of Energy (US DOE) operated a loan/lease program for Californium-252 (^{252}Cf) sealed sources from the late 1960s until 2008. The ^{252}Cf Loan/Lease Program was established to provide low-cost access to Cf-252 sealed sources for educational, research, and medical applications performed by academic and government research institutions. Sources were loaned for a period of between 1 and 5 years, and loan contracts were renewable at each customer's discretion. The sources were most frequently used as neutron sources; a sealed source containing a single microgram of ^{252}Cf emits 2.3 x 10^6 neutrons/second due to spontaneous nuclear fission [1]. Sources used in the program contained micrograms to milligrams of ^{252}Cf in oxide form when newly fabricated. Loan program customers used these sources for various purposes including neutron radiography, well logging, nuclear instrument calibration, nuclear reactor start-up, cancer therapy research, and prompt gamma neutron activation analysis. Although many sources were loaned to a single customer for the duration of the loan program, some sources were loaned to multiple customers over time. As the sources aged, lower neutron flux ranges could be applied to different research and development applications.

Program sponsorship ended in 2008 when DOE discontinued the program with no provision for follow on programmatic sponsorship. In total, 467 sources remained in the program inventory when the program was discontinued with 255 sources held in storage at Oak Ridge National Laboratory (ORNL) and 212 on loan to 52 academic and government research entities[1]. This placed the sources within the purview of the Office of Nuclear Material Integration (ONMI) for reassignment, recovery, reuse or disposal. [2].

To provide for safe, effective and economical final stewardship of these sources, ONMI initiated the ^{252}Cf Loan/Lease Program Closeout Project in 2013. The purpose of this project was to close all loan customer accounts through reassignment of the source or return to DOE for potential reuse or disposal. The program was successfully completed in 2019, and involved investigative work as well as resolution of technical challenges. The lessons learned from this program may be beneficial for entities that must recover legacy radioactive materials distributed to academic and government research institutions and/or commercial enterprises by programs, projects or initiatives that have been discontinued.

II. PROGRAM PHASES

A. *Re-establishing Communications with Customers*

One key feature in re-establishing communication with the customers was a decision to take a collaborative approach in the return of these sources. Customers were treated as partners in the closure effort, with costs to them reduced as much as possible and flexible scheduling to fit their needs. By maintaining communication with loan customers, the sources current condition and storage configuration could be updated. Not only was this a critical step in verifying the current location and condition of the sources, but it established confidence within the customer base, many of whom were concerned about the uncertain future of the ^{252}Cf sources in their possession.

Due to the amount of time between when the program was discontinued (2008) and the Closeout Program was established (2013), critical records needed to support the project had to be located. This included information concerning the radioactive content, past shipping and receiving information, and any visual information such as drawings, sketches, or pictures.

B. *Determining a Disposition Path*

There were two potential disposition paths for each loaned source. The source could remain with the customer or the source could be returned to Oak Ridge National Laboratory (ORNL, the project's original headquarters) for final disposition at the Waste Isolation Pilot Plant (WIPP). The sources loaned to other DOE-owned facilities were transferred to the custody of an active DOE program[2]. For sources loaned to academic and other government research institutions the choice was usually to return the source for final disposition.

Due to the presence of alpha decay product Curium-248 (^{248}Cm) and the radioactivity of ^{252}Cf, the loaned sources are

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[1] Only the sources on loan to the 52 academic and government research entities will be discussed in this paper.

[2] Program managers were also amenable to transferring ownership over to other US government entities.

considered to contain transuranic (TRU) isotopes. DOE facilities authorized to ship TRU waste to WIPP) were able to incorporate the item(s) into their site specific shipping schedule. Academic and commercial institutions that did not have this option needed to return the material to ORNL for terminal management and disposal. An alternative those institutions had was to assume ownership and responsibility of the source under an NRC license. Customers who choose this option were required to register their sources with the DOE/NNSA Off-Site Source Recovery Program (OSRP[3]) for tracking and eventual disposal as TRU waste.

C. Source Retrieval Method

If a source was no longer desired at the facility and was in need of retrieval, the facility had two options; self-return or team retrieval by ORNL (shown in Figure 1).

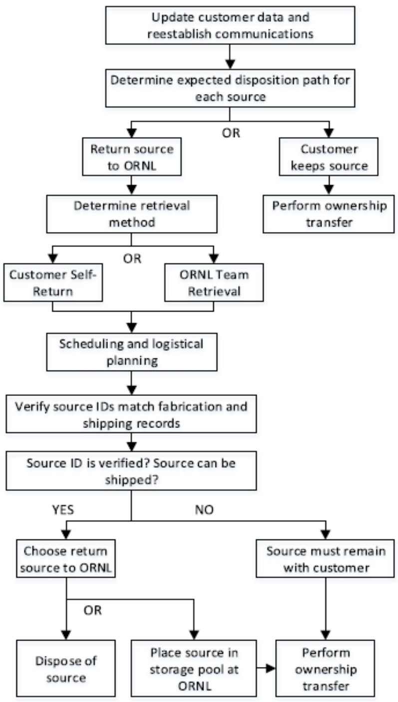

Figure 1. Block flow diagram of disposition pathways for loaned sources

In the case of a self-return, the facility in possession of the ^{252}Cf source received a shipping container(s) from ORNL, loaded the source(s) into the container and shipped the container back to ORNL. This option required the facility to have staff qualified to perform this task and for the facility to be shipper of record. The "shipper of record" is responsible for regulatory compliant delivery of the package to the receiver. In the U.S., the shipper of record is responsible for any accident associated with transport. In cases where ORNL retrieved the source, a team of certified transportation specialists were sent to the institution to package and retrieve the source(s). Under this option, ORNL acted as the shipper of record.

D. Logistics

One of the most complex aspects of the program was schedule management and logistics, due largely to the finite number of available shipping containers for returning the sources. Compounding this, ORNL only had one retrieval team available and certain facility locations had restrictions regarding when their sources could be accessed. All of these factors needed to be addressed to ensure timely and cost effective source recovery.

Additionally, shipping records are required to accurately describe the shipments contents. Therefore, the identity of each source at the institution had to be verified with ORNL's records prior to shipment. This identification step, helped mitigate the risk of loading an unspecified source into the shipping container when returning their ^{252}Cf source(s).

E. Disposition of Returned Sources

All sources returned to ORNL were either put in the WIPP disposal queue upon receipt or, if they contained more than 1,000µg ^{248}Cm, placed into a storage pool for recovery of the ^{248}Cm. ^{248}Cm is a rare isotope used to create analytical radiochemistry standards and support heavy element research missions. The 1,000µg cutoff was established as a cost-effective measure; due to the cost of extracting ^{248}Cm from a sealed source.

III. PROGRAM EXECUTION

A. Packaging Types

The loaned ^{252}Cf sources are categorized as special form radioactive material under the Department of Transportation's Title 49 of the Code of Regulations (CFR) Part 7. Special form radioactive materials are encapsulated sources which only pose an external radiation hazard, not a contamination hazard if the package is damaged or ruptured during transport. Special form materials can be shipped in what is known as "strong tight" or Type A packaging with the radiation exposure risk being attenuated with auxiliary equipment. ORNL owns eleven Type A shipping casks that are regularly used to support ongoing ^{252}Cf mission work. The shipping casks are heavy, weighing up to 1,444kg, and require a crane for handling. They can accommodate up to 3,700 µg ^{252}Cf. Loan lease program customers were not expected to handle these large shipping casks on their own and would require use of the single available

[3] OSRP is a program funded by DOE, through the Los Alamos National Laboratory (LANL), and retrieves sources from commercial sector groups and state agencies in possession of at-risk sources.

ORNL retrieval team. Moreover, only four out of these eleven shipping casks were made available to support this recovery effort. The sources on loan to customers contained between 0.02 and 15 µg of ^{252}Cf, meaning 95% of them could be transported in smaller, lighter-weight Type A containers if available.

The program needed to make a decision that would relieve pressure on customers and the retrieval team while adequately meeting the demand schedule. The program created a Type A shipping container that weighed 59 kg, provided adequate shielding to attenuate radiation dose, and could be moved with a hand cart, eliminating the cost and risk of using a crane. Seven of the newly designed Type A packages were fabricated at a cost of $3,000 each.

Now that safety risks were significantly reduced, the program encouraged institutions to pursue the self-return option, which reduced the burden on ORNL manpower as well as allowed the institution to ship on their own schedule. Furthermore, these lighter weight containers allowed the program to use a common commercial carrier for shipments.

By limiting the number of ORNL team retrievals, the program was able to better support recovery of sources requiring the larger, heavier package, or institutions who did not have personnel with the proper qualifications and expertise to compliantly ship radioactive material. This permitted the program to create an efficient system for staging the retrieval of sources through regional recovery campaigns. Executing regional retrieval missions allowed the program to achieve significant cost efficiencies.

IV. Results and Discussion

The first two years of the project were spent preparing the institutions for successful shipment or retrieval[4]. Table 1 shows the number of remaining sources and institutions by year. Table 2 shows the disposition path count for each source by year.

TABLE I. SOURCES ON LOAN BY YEAR

Year	Remaining number of sources on loan, per year	Number of loan customers still holding loaned sources, per year
2012	212	52
2013	208	49
2014	194	44
2015	192	41
2016	150	36
2017	33	15
2018	1	1
2019	0	0

TABLE II. RETURNS OVER PROGRAM DURATION

Year	Self-Returns	Team Returns	OSRP	OT[5] to DOE	OT to NRC
2012	0	0	0	0	0
2013	0	0	4	0	0
2014	14	0	0	0	0
2015	2	0	0	0	0
2016	0	24	0	7	11
2017	9	15	0	92	1
2018	14	17	0	0	1
2019	1	0	0	0	0
Total	40	56	4	99	13

Large cost savings were created by encouraging facilities to select the self-return option, by setting up efficient retrieval combination missions, and by working directly with the site to transfer ownership when appropriate.

Transferring ownership of the sources was the preferred option because for the most part, the sources would continue to be productively used rather than dispositioned. In total, ownership of 112 sources was transferred; 99 to DOE-regulated sites, 12 to Nuclear Regulatory Commission (NRC) regulated sites, and one to a foreign facility.

The Legacy ^{252}Cf Sealed Source Closeout Project addressed the loans of 212 sources at 52 academic and government research entities. All loans were closed in just six years, with a total cost of $1.3 million. The innovations developed and implemented through this project created a savings from the original cost and schedule estimates of 9 years and $2.5M.

V. Lessons Learned

Future undertakings involving source retrieval can apply the lessons learned from the Legacy ^{252}Cf Sealed Source Closeout Project, they are:

- It is vital that the customer be involved in planning and executing the source retrieval. Problems often cannot be imposed on the customers and solutions regularly need to be coordinated with other federal government agencies, as well as, external entities. To ensure a well-managed, cost effective and safe recovery, early customer involvement is a must.
- Each source must have a complete data package that matches source records to address transportation requirements. This also gives the receiving facility confidence that the source can be effectively and safely managed by the receiver.
- Disposition paths must be identified before taking any action to move the source(s). If there is no known disposition pathway, an analysis of alternatives or options analysis using relevant subject matter expertise may be required to identify the most appropriate disposition pathway(s).

[4] During the communication stage, it was discovered that four sources had already been returned using OSRP.

[5] OT is Ownership Transfer

- While self-returns can contribute to project efficiencies, they may require more time to accomplish, due to conflicting priorities at the shipping facility. These delays can be avoided by working closely with the facility to ensure that the return timeline is coordinated with that of the receiving facility.
- Labor availability and logistics will likely control the schedule. If sources require a specialized retrieval team, the schedule will be driven by team availability. If customers perform self-returns, the schedule will be driven by shipping container availability and the schedule for packaging and shipping at the shipping location. In either case, if shipping containers are limited in supply, the schedule will be affected by how quickly the receiving location unloads the shipping containers so they can be sent out again. All schedules must be highly coordinated.

VI. CONCLUSIONS

Through careful planning and attention to efficiency and cost effectiveness, DOE was able to close out the loan accounts on 212 ^{252}Cf sources from 52 facilities, as well as, disposition of 255 sources held in the ORNL storage pool. Disposition pathways for the sources included retrieval, transfer to an active DOE program, or transfer to an NRC licensee. Lessons learned during the closeout effort include: detailed research and retrieval of source specifications and verification of this information; sizing of transport packages to encourage institutions to self-return; ensuring the availability of trained and qualified staff to execute packaging and transportation of sources; incorporation of a regional recovery strategy to expedite recoveries and minimize cost; and careful and deliberate scheduling of recovery efforts to harmonize facility and recovery team schedules.

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Coupled Thermal, Hydrological and Chemical Model for Transport of U(VI) in Engineered Barrier System and Host Rock for a Generic Nuclear Waste Repository Case

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I. INTRODUCTION

In the context of nuclear waste management, performance assessment (PA) is "the process of quantitatively evaluating the ability of a disposal system to contain and isolate radioactive waste".[1] This evaluation is further used for the design and development of the terminal nuclear waste repository and will determine compliance with applicable regulations.[1] Most of the requirements are probabilistic since many conditions are hard to predict and impossible to test and observe empirically due to the prolonged time horizon of up to 1 mln years. As a result, PA gives an estimate of the likelihood of each process and event that is coupled with different potential scenarios of the future of the disposal system and their effects on the system. Since such calculations involve a certain level of uncertainty, uncertainty analysis is a part of PA.

Thermal, Hydrological, Mechanical, and Chemical (THMC) processes impact flow parameters due to thermal processes and buffer swelling, and geochemical parameters such as clay mineral and *pH* changes within or near engineered barrier system (EBS) or bentonite.[2,3]

The development of a generic performance assessment (PA) has the challenge of integrating THMC/THC into the overall PA model. The uncertainty associated with these processes is not accounted for in the overall PA and poses a great challenge for the integration of multiple coupled processes models into the global PA model. Without this integration, it is impossible to identify important parameters for reducing the overall uncertainty (OU).

Here the methodology to represent the whole buffer grid through the distribution coefficient, Kd, to integrate the EBS THC model into the PA model is presented. Kd is an important parameter to estimate the mobility of contaminants in aqueous solutions in contact with the surface, subsurface and suspended solids, and calculated as a ratio of solid phase to solute concentrations. Thus, Kd is able to describe the transport of radionuclides in EBS as a single variable in overall PA.

For this purpose, the model assumes that the host rock has the properties of well-characterized Opalinus Clay [4,5], as well as bentonite, which is assumed to be FEBEX bentonite, which has also been extensively characterized by laboratory and field experiment. [6]

II. MODEL DEVELOPMENT

A. Conceptual model

TOUGHREACT V3.3-OMP, multiphase fluid flow and reactive transport simulator [7] is used to simulate the processes in the conceptual model. In our case, TOUGHREACT is applied to one-dimensional porous and fractured media. We developed a numerical model to explore the impact of the interaction between host rock and buffer from 1000 to 100 000 years assuming that at that point the canister will be fully corroded.

As a result, we also consider two-phase (liquid and gas) flow in the model due to the presence of vapor diffusion of the water flow in the bentonite barrier and host rock during the early unsaturated stage. [8,9] 95% of the spent nuclear fuel rods consist of ^{238}U (half-life, $t_{1/2}$=4.468×10^9 years)[10] and after releasing from the repository it undergoes several oxidation processes and can be transformed from insoluble and immobile *U(IV)* to highly mobile *U(VI)*. Therefore, for the purpose of this work we focus only on the migration of *U(VI)* to be conservative and assume that after degradation of UO_2 by dripping water all high-level waste consists only from ^{238}U (schoepite phase). [11-13]

The following reaction equation (1) allows us to control the concentration of *U(VI)*

$$Schoepite + 2H^{+} = 3H_2O + UO_2^{+2} \quad (1)$$

The important reaction of adsorption/desorption controls the migration of *U(VI)* in bentonite. The reactive transport model we are building utilizing the distribution coefficient, Kd approach. Kd is a measure of sorption of contaminants to soils and is defined as the ratio of the quantity of adsorbate adsorbed per unit mass of solid to the amount of the adsorbate remaining in solution at equilibrium (2).

$$Kd = \frac{Mass\ of\ Adsorbate\ Sorbed}{Mass\ of\ Adsorbate\ in\ Solution} = \frac{A_i}{C_i} \quad (2)$$

where $A_i = C_i + A$

A= free or unoccupied surface adsorption sites

C_i = total dissolved adsorbate remaining in solution at equilibrium

A_i = amount of adsorbate on the solid at equilibrium. We use this parameter to describe the *U(VI)* retardation in host rock and bentonite.

B. Numerical model

We have two zones: bentonite buffer and host rock (BHR).[2] Initial conditions across the whole BHR: temperature is uniform and equal to 12ºC. It peaks within 10-100 years at ~100ºC near canister and after 1000 years it decreases to ~18ºC while the temperature at the very last grid block at 50 m stays almost the same ~12ºC; bentonite and host rock are fully saturated after 1000 years. To demonstrate the THC changes in BHR, we mostly used the temporal evolution at points A, B, C, and D located on the bentonite and host rock (Fig. 1).

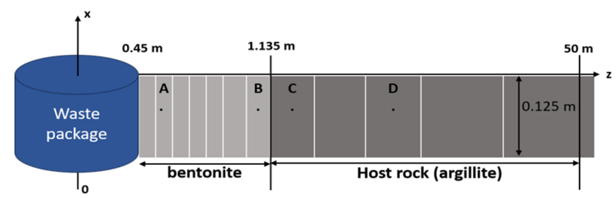


Figure 1. Model mesh, not to scale: Point A is located at r=0.479 m; B is at r = 1.13 m in bentonite; C is located in host rock next to the BHR interface at r=1.3 m, and D is 10 m away from the BHR interface.[14]

C. Sensitivity Analysis

To evaluate which parameters of the system have the highest impact on *U(VI)* transport, we investigate the impact of each parameter on each of the simulation outputs. We performed additional simulations changing ten chemistry parameters from TABLE I. We use the so called two-site protolysis nonelectrostatic surface complexation and cation exchange sorption model.[15] Surface protonation reactions involve a strong site (ill_sOH or sme_sOH) and two weak sites (ill_w1OH, ill_w2OH or sme_w1OH, sme_w2OH) are used to describe acid-base titration measurements on montmorillonite and illite.

TABLE I. PARAMETERS CHANGED DURING SIMULATION

Parameters	***Range***
Site density (cm2/g): ill_sOH	10^{-6}-10^{-4}
Site density (cm2/g): ill_w1OH	10^{-6}-10^{-4}
Site density (cm2/g): ill_w2OH	10^{-6}-10^{-4}
Site density (cm2/g): sme_sOH	10^{-6}-10^{-4}
Site density (cm2/g): sme_w1OH	10^{-6}-10^{-4}
Site density (cm2/g): sme_w2OH	10^{-6}-10^{-4}
Volume fraction: calcite	10^{-3}-10^{-1}
Initial pore water composition: H^{+}	$1.91*10^{-9}$
Volume fraction: montmorillonite	0.3-0.95
Volume fraction: illite	0.01-0.2

III. RESULTS AND DISCUSSION

A. Temporal evolution

The chemical conditions, especially within the buffer, play a significant role in schoepite dissolution and aqueous complexation with *U(VI).* Some studies have shown that HRB interaction affects the geochemical evolution within the bentonite barrier.[16]

In the model, total *U(VI)* in bentonite is described as aqueous, exchanged, and adsorbed phases. Based on the simulated *U(VI)* concentrations in the aqueous and solid phases, we computed the Kd values at each grid block. The results of the temporal evolution of *U(VI)* concentration and Kd from simulations are presented in Fig. 2. Chemistry parameters were changed (see legend for Fig. 2 a) and TABLE I. The clay main components, smectite and calcite, affect the *U(VI)* transport the most: this is due to the presence of Ca^{2+}, that reacts with schoepite release from the waste package and due to the fact that roughly 80% of adsorbed *U(VI)* is on smectite by the end of the simulation time.[3] Although the concentration of *U(VI)* increases in these cases it does not reach a background level of *U(VI)* = 1.26×10^{-6} mol/kg.[17]

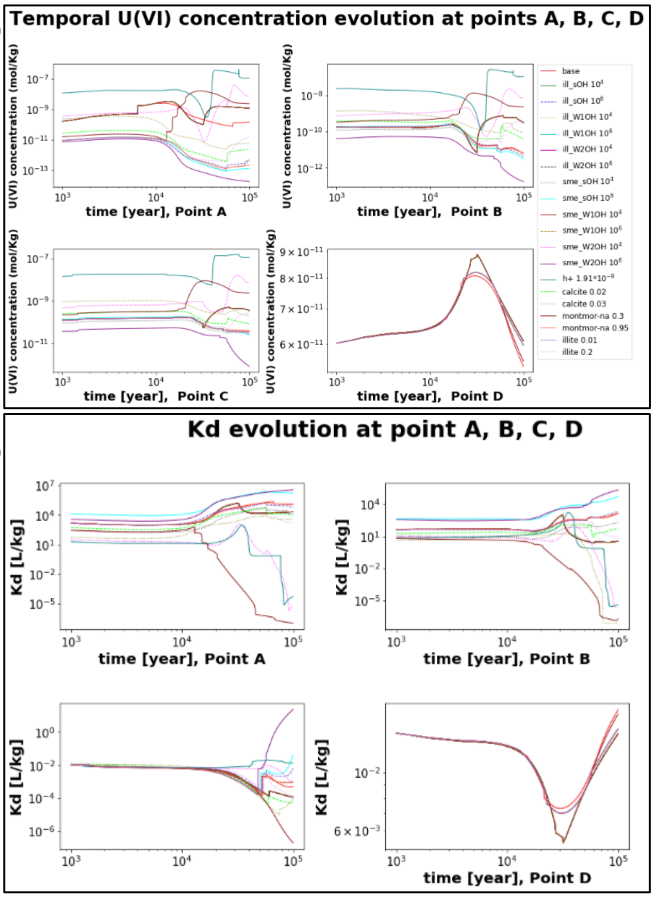
a)
b)


Figure 2. Simulated temporal evolution of a) *U(VI)* concentration and b) Kd at points A-D over time.

Kd spatial averaging follows the general trend of the temporal evolution of Kd at points A and B. The weighted average Kd values are relatively similar to the maximum spatial values. Overall response of Kd to chemistry changes is strong which makes Kd a perfect candidate to represent the whole EBS grid block in PA model without loss of information.

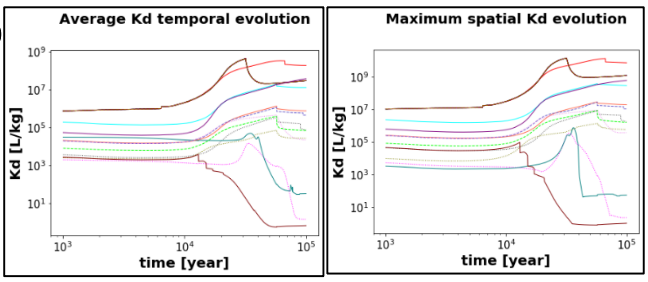
a)
b)


Figure 2. a) Simulated weighted mean spatial Kd over time and b) simulated maximum spatial Kd over time.

In most cases, Kd and pH have linear correlation and in 100 000 years pH does not become lower than 6.5

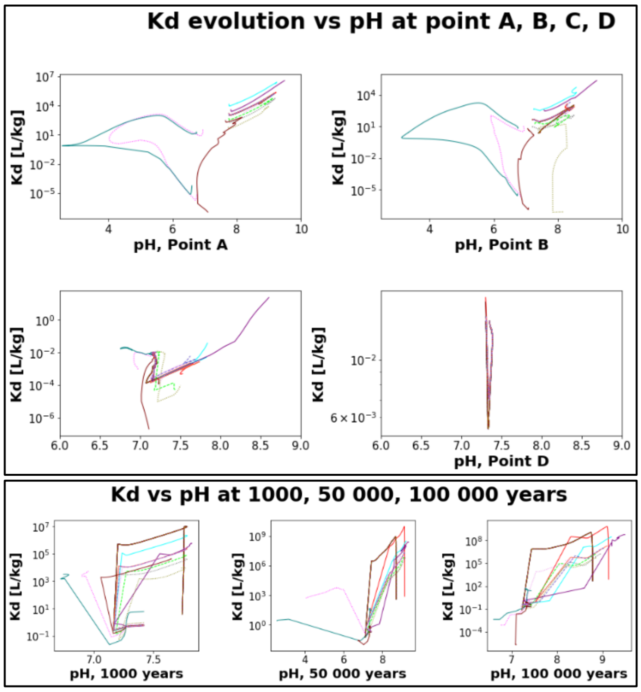
a)
b)


Figure 3. a) Simulated evolution of Kd *vs pH at points A-D, b) simulated* Kd *vs* pH *in 1000, 50 000, 100 000 years.*

The results shown let us assume that Kd is an important parameter for radioactive transport. It is commonly used to define the retardation factor, the ratio of the average linear velocity of water divided by the average linear velocity of the contaminant. [18]

IV. Conclusion and Future Work

Here we presented a methodology that has a potential to integrate the THC model with the PA model through Kd. First, we simulated the THC process within an EBS-HR (base case) system using TOUGHREACT and computed Kd. Second, we changed chemical parameters that may affect the evolution of Kd and performed more simulations to compare changes with the base case and assess their influence on Kd. We were able to assess the spatial and temporal evolution of Kd The spatial averaging has the potential to represent the entire buffer grid block at the PA model, however, additional sensitivity and principal component analysis are required and will be performed in future work.

Kd vs pH dependency is fairly linear, and this dependency might be used for reduced-order model development in future work, although more simulations need to be performed.

In further works it also necessary to perform simulations for other actinides with half-life > 1000 years.

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Dismantling and Management of Activated Concrete arising from the Decontamination and Decommissioning of a Radiopharmaceuticals Production Facility in Belgium

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I. Introduction

On May 2012, Best Medical Belgium S.A., a manufacturer of pharmaceutical products located on the nuclear site of Fleurus, was declared bankrupt. According to the Belgian law, ONDRAF/NIRAS (Belgian Radioactive Waste Management Agency) was entrusted with the clean-up and the Decontamination and Decommissioning (D&D) program of the former facilities, nowadays known as "ONDRAF Site Fleurus" (ONSF). In June 2015, ONSF launched the D&D program, for which Tractebel supports the definition of the clean-up and D&D strategies; the writing of the Final Decommissioning Plan (FDP) & License Application for decommissioning; and the follow-up of the upcoming D&D activities on site.

Four nuclear buildings are part of the D&D activities. One of them, the Building 14 (B14), represents the most challenging task with the clean-up of highly contaminated areas and its further decommissioning up to the greenfield. Beside its two cyclotrons successively used for the production of radioisotopes (^{201}Tl, ^{123}I, ^{18}F, etc.) this building hosts hot cells and glove boxes used for the purification of the radioisotopes and the conditioning of the final pharmaceutical product.

The B14 presents a significant amount of activated concrete, part of the ceilings, walls and floors of the shielded cyclotrons and targets rooms. This amount of activated concrete will constitute the most important part of radioactive materials arising from the D&D activities.

A characterization program has been launched to define the activation depth of concrete. Based on these results, the dismantling plan of concrete has been defined in a safe and efficient way in order to guarantee the building stability, the confinement of radioactivity during the operations, as well as the reduction of the radioactive waste amount.

Figure 1. IBA Cyclone® 30 cyclotron within its shielded room

II. Inventory of the Building 14

In order to prepare the clean-up and the D&D studies and works, a physical inventory of the relevant installations has been drawn up by Tractebel in the early stages of the project. The B14 consists of four main nuclear zones (Figure 2) :

- The "CGR zone" where one CGR 70 MeV cyclotron that generated radioisotopes is located;
- The "IBA zone" where one IBA Cyclone® 30 cyclotron (Figure 1) that generated radioisotopes is located;
- The "Chemical zone" where some hot cells and glove boxes for the purification of the radioisotopes generated by the both cyclotrons are located;
- The "$^{90}Sr/^{90}Y$ zone" where some hot cells and glove boxes for the production of pharmaceutical products are located.

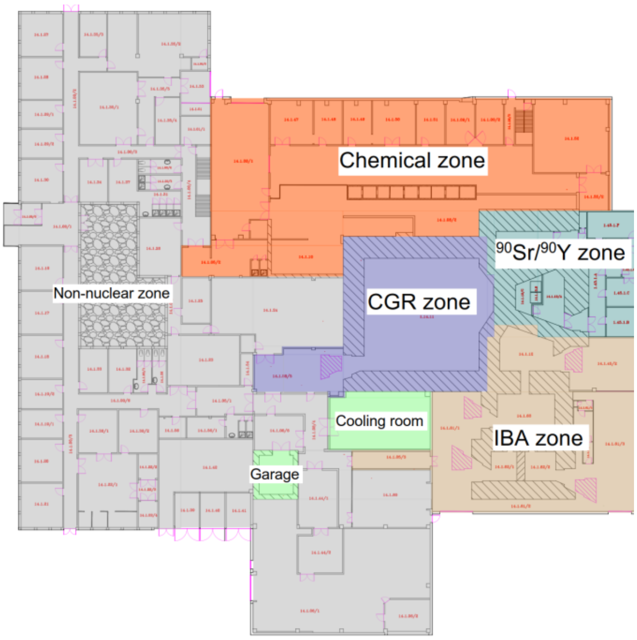

Figure 2. Representation of the zones within the building 14

Other nuclear zones exist within the B14, amongst them the cooling room where the cooling systems of both cyclotrons are located and the garage where the repairs of some specific cyclotron pieces were performed. The other part of the B14 consists of a non-nuclear zone for the offices, the cold workshop, the sanitary, etc. The specific mass inventory of the B14 is provided in Table I.

TABLE I. MASS INVENTORY OF THE BUILDING 14

	Mass inventory [kg]	
	Suspected activated	***Suspected contaminated***
Equipment	330,602	496,803
Infrastructures	10,815,625	2,941,907

III. ACTIVATION DEPTH OF CONCRETE

In the period from 1993 to 2012, the CGR and IBA cyclotrons have been successively used for the production of radioisotopes by the irradiation of targets with proton beams. When the protons impacted on targets, neutrons were produced via spatial reactions, which, in turn, caused the activation of the surrounding infrastructures of the shielded cyclotrons and targets rooms.

First, Tractebel has developed a theoretical characterization model to estimate the neutron flux distribution within the facility in order to identify possible hot spots, as well as to estimate the activation depth of the surrounding infrastructures (ceilings, walls and floors). The model enables to quantify the neutron fluxes by using the Monte Carlo code MCNPX2.7.0. and the nuclear data library ENDF/B-VII.1, considering a realistic description of the facility.

Based on the theoretical results of activation depths, Tractebel has performed a radiological characterization campaign of the B14 in order to increase the level of details regarding the activation of the infrastructures. The campaign consists in sampling the ceilings, walls and floors of the CGR and the IBA zones. The results obtained by the theoretical model aim to define the parameters of the on-site characterization campaign such as the number of samples and their associated location and depth. It is worth to note that the sample locations have been determined as much as possible in a conservative logic, taking into account the layout constraints.

In a first iteration, the following assumptions have been considered:

- The same activation depth is taken for one entire infrastructure type;
- Some samples have been taken on the reinforced bars located into the concrete in order to determine their activation. Nevertheless, this has not been taken into account to determine the activation depth of the concrete.

A second characterization campaign could refine the estimations if it is judged necessary when developing the management strategy for the activated concrete.

A comparison between the theoretical and the measurements results has been performed in order to validate the MCNPX model. The specific activities of ^{60}Co and ^{152}Eu, amongst others, of samples have been measured for this comparison. Indeed, these two isotopes constitute the major contributors to the dose incurred by the workers. On average, the agreement between calculations and measurements is satisfying, with ratios calculations/measurements typically between 0.5 and 2, with few outliers, where a factor 5 or more can be observed. These differences are larger than the error of measurements. However, considering the number of uncertainties inherent to this type of studies, this can be considered as acceptable and the proposed MCNPX model has been validated.

IV. MANAGEMENT OF ACTIVATED CONCRETE

The operation of both cyclotrons generated a large amount of slightly activated concrete. The strategy for its management has been defined to reduce the amount of radioactive waste and to be cost-effective. It is worth to note that immediate dismantling is the reference scenario to establish the D&D program of the B14.

To reduce the amount of radioactive waste, the conditional free release has been considered as an available disposal route, in agreement with the Belgian Authorities. The strategy considers the following removal routes for the management of dismantled activated concrete (Figure 3):

- The activated concrete with a level of activity higher than the exemption limits [1] is considered as radioactive waste and will be, after dismantling, transferred to ONDRAF/NIRAS for further final disposal;
- The activated concrete with a level of activity lower than the exemption limits but higher than the clearance limits [1] is considered for conditional free release and will be transferred, after dismantling, to an industrial landfill for monitored storage;
- The rest of the concrete for which the level of activity is lower than the clearance limits is considered for unconditional clearance.

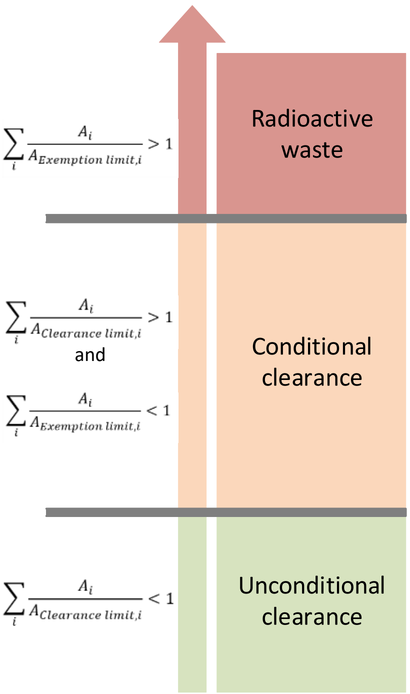

Figure 3. Strategy considered for the management of activated concrete (A_i, activity concentration of radionuclide i and $A_{Clearance/Exemption\ limit\ i}$, the associated clearance/exemption limit)

The main depth distribution of activated concrete between the conditional clearance and the radioactive waste is illustrated on Figure 4, for the specific case of walls. The analytic results are provided in Table II.

TABLE II. DEPTH DISTRIBUTIONS OF ACTIVATED CONCRETE BETWEEN THE CONDITIONAL CLEARANCE AND THE RADIOACTIVE WASTE

Depth of infrastructures [cm]	Conditional free release	Radioactive waste
CGR zone	20 to 50	-
IBA zone	20 to 140	20 to 40
$^{90}Sr/^{90}Y$ zone	30 to 50	-

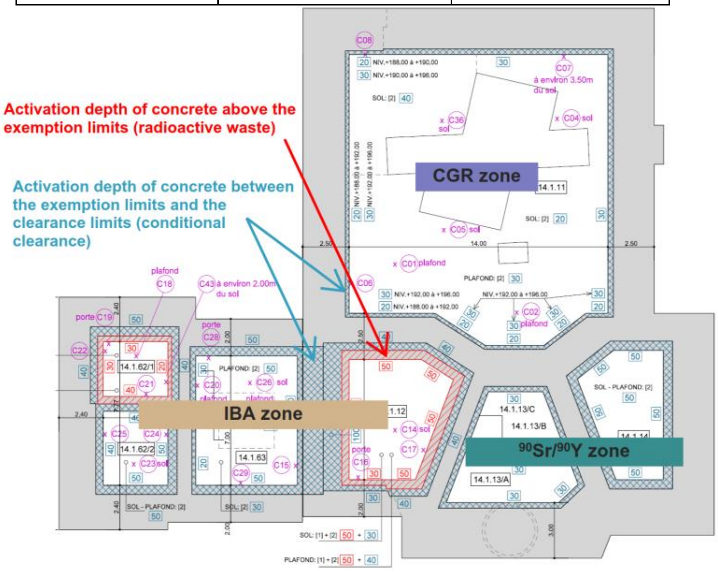

Figure 4. Activation depth of concrete within the CGR, IBA and $^{90}Sr/^{90}Y$ zones (specific case of walls)

V. DISMANTLING OF THE ACTIVATED CONCRETE - EXTENSION OF THE EXISTING BUIDLING 14

The activation depth of concrete within the CGR zone enables its dismantling from the inside of the shielded room, without jeopardizing the stability of the infrastructures. The activation depth is bigger within the IBA zone which complicates the dismantling of the activated concrete.

Two options have been identified for the dismantling of the activated concrete within the IBA zone:

- The removal of the activated concrete is performed from the inside of the existing IBA zone;
- The removal of the activated concrete is performed from the outside of the existing IBA zone.

Most of the ceilings, walls and floors are partially activated. One wall between two shielded targets rooms from the IBA zone is activated in its complete depth, as illustrated on Figure 4, and will thus have to be entirely removed. The stability of the building must be ensured during all of the dismantling activities. The removal of such depth of concrete from the walls, leaving the ceilings in place, will jeopardize the stability of the infrastructures, cause additional risks and complicate the realization of the dismantling operations. Furthermore, the removal of the activated concrete of the ceilings from the inside of the existing IBA nuclear zone will generate additional risks regarding the conventional security of those potentially working under the infrastructures in dismantling.

To decrease the conventional risks and to simplify the dismantling operations, it has been decided to remove the activated concrete from the outside of the existing IBA zone. The strategy considers two phases:

- Phase 1 consists in the removal of all the ceilings of the IBA and $^{90}Sr/^{90}Y$ zones, comprising the activated and the non-activated parts of the concrete, each block being cut in the entire height of the ceilings. This will provide the necessary access to the walls and floors for their further dismantling (phase 2). The dismantled concrete block will then be managed in another location to be sorted in function of the considered removal route;
- Phase 2 consists in the removal of the activated part of the walls and floors, the free releasable concrete remaining in place in this case. The dismantled concrete blocks will then be managed in another location to be sorted in function of the considered removal route, if needed.

This strategy has nevertheless the disadvantage of losing the confinement of radioactivity. To guarantee this confinement during all the dismantling phases, a new building will be erected above a part of the IBA zone, namely the confinement of the IBA zone (Figure 5). It is worth to note that the other part of the IBA and $^{90}Sr/^{90}Y$ zones are already covered by an existing room (Figure 5).

To support the management of dismantled concrete blocks, the confinement will be extended in line with the IBA zone. This extension will be equipped with the necessary functionalities:

- A confined cutting box to sort the dismantled concrete blocks between the considered removal routes (unconditional clearance, conditional free release and radioactive waste). This also aims to meet the requirements of the associated removal routes;
- A packaging station to pack the final concrete blocks into plastic;
- A radiological characterization room, with a low background, to perform the radiological measurements;
- A filling station to fill the IP-2 ISO 20’ containers with the dismantled concrete blocks for their evacuation from the site;
- Two cranes to handle the dismantling concrete blocks during all the phases of their management: one crane covering the existing IBA and $^{90}Sr/^{90}Y$ zones and another one covering the extension of the IBA zone. The reason to consider two cranes is to make a separation between the confinement and the extension, the confinement allowing the presence of contamination (such as activated dust from cutting of concrete blocks) and the extension being considered as a clean zone. In addition, it allows also to perform the in-situ dismantling operations in parallel with the management of the dismantled concrete blocks (sorting, packaging, characterization and filling into ISO 20’ containers);
- All the necessary auxiliary systems such as the ventilation system, the radiation monitoring systems, the electricity, the compressed air, the water, etc.

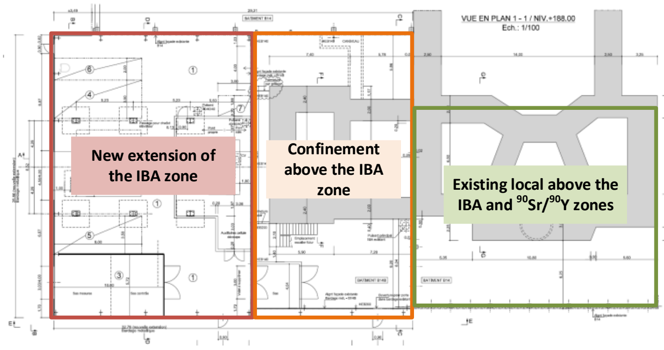

Figure 5. The future confinement and extension building to support the D&D activities for activated concrete

The dismantling of the activated concrete from the CGR, IBA and $^{90}Sr/^{90}Y$ zones is currently planned to be executed starting from August 2022 until December 2025.

VI. Conclusions

A large amount of slightly activated concrete will be generated during the D&D activities of the B14. A theoretical model has been developed to provide a preliminary assessment of the quantity of activated concrete. The results of the MCNPX model have been validated by an on-site characterization campaign, taking samples from the B14 infrastructures.

The conditional free release has been considered as an available removal route for the management of a part of the activated concrete, the other part being considered as radioactive waste. This strategy has been validated by the Belgian Authorities and is cost-effective due to the reduction of the amount of radioactive waste. The activated concrete meeting the radiological limits for the conditional free release will be transferred to an industrial landfill for monitored storage.

To decrease the conventional risks and to simplify the dismantling operations, the dismantling of activated concrete will be realized from the outside of the existing IBA nuclear zone. To guarantee the confinement of radioactivity during all the dismantling phases, a new building, named the confinement, will be erected above the IBA zone. This will be extended to support the management of dismantled concrete blocks (sorting, packaging, characterization and filling into ISO 20’ containers).

VII. Reference

[1] Royal Decree of the 20th of July 2001, “General regulations for the protection of the public, the workers and the environment against ionizing radiation”

Uranium Processing in Australia – Past, Present and Future

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I. Introduction

The processing of uranium in Australia has a long but relatively limited history, with only nine uranium mills having operated since 1954, and only three of these still producing uranium today [1]. While social and political factors have impacted the development of uranium deposits over the last 60 years, other factors including economic, technological and environmental factors have had as big a bearing on the number and type of uranium mines operating in Australia today.

II. Types of Uranium Mining and Processing

There are three main mining processes used to extract uranium minerals from mineral deposits, namely open-pit mining, underground mining and in-situ leaching (ISL, also known as in-situ recovery). Open-pit mining is common practice when the deposit is located relatively close to the surface, generally to a depth of up to 200 m [2]. Open-pit is preferable to underground mining as it generally achieves better ore recoveries, has higher productivity and has safer mining conditions for workers [2]. Underground mining is used where open-pit mining is unsuitable due to the depth of the deposit or where open-pit mining is otherwise unviable [3]. Underground mining involves more precise methods, known as stoping, to target higher grade ores within the deposit. Compared to open-pit mining, underground mining produces significantly less waste rock and has a significantly smaller surface footprint [4]. ISL mining involves extracting uranium from typically low grade deposits located in confined underground aquifers by pumping an aqueous solution from the surface to extract the uranium from its natural mineral [5].

The flowsheet for processing uranium-bearing ore obtained from open-pit or underground mining has not changed significantly in the last 60 years. The flowsheet typically involves ore crushing, grinding, leaching via sulfuric acid or alkaline pathways (in some cases a heap leaching process is used, as operated at Nabarlek and recently planned by Olympic Dam), solid/liquid separation, solvent extraction (SX) or ion exchange (IX), chemical precipitation, and drying/calcination to produce the end product uranium oxide concentrate (UOC) [2]. The flowsheet for uranium processing via ISL, once the uranium is extracted from the deposit, is identical and typically involves SX or IX, chemical precipitation and drying/calcination [5].

The selection of the mining/processing route for a uranium deposit is typically made on the basis of: 1) the mineralogical form and grade of the deposit; 2) the geological structure in which the uranium mineral resides; 3) the degree to which uranium can be extracted/recovered; 4) the type of mining permit granted to the mining company; 5) environmental impact considerations such as surface disturbance or potential for ground-water contamination; and 6) climate conditions including water availability and quality.

III. History of Uranium Processing in Australia

The first uranium deposits mined in Australia were located at Radium Hill and Mount Painter in South Australia. Operating intermittently up until the 1930s, these mines were operated to recover their radium content, which was predominantly used in the medical industry [6]. After the Second World War, uranium exploration in Australia took a major turn upwards to support military requests from the Western Allies. This exploration program discovered deposits such as Rum Jungle in the Northern Territory and Mary Kathleen in Queensland, which went on to produce the most significant quantities of uranium in Australia in the 1950-60s [6].

Including these mines that operated in the 1950s, only nine uranium mills have ever operated in Australia. From 1954 to 1971, five uranium mills produced over 9,000 tonnes of UOC, which was only a small fraction of the global production in this period [7]. The decreased demand in uranium for defence-related purposes saw the closure of several of these mines in the early 1960s and a near-cessation of uranium exploration in Australia for a few years [8].

The late 1960s brought about a major turn in the Australian uranium exploration and mining industry, where a new governmental export policy enabled exploration companies with large budgets to scour the country for bigger and higher grade uranium deposits [9]. This exploration effort discovered enormous deposits including Olympic Dam, Beverly and Honeymoon in South Australia and Ranger, Jabiluka, Koongara and Nabarlek in the Northern Territory [9]. These exploration efforts contributed significantly to establishing Australia as the most uranium resource rich country in the world.

The early 1980s saw the introduction of a Government policy that restricted uranium production to just three mines, namely Ranger, Olympic Dam and Nabarlek [9]. With Nabarlek finishing production in the late 1980s, Australia did not produce uranium from any new mines until the start-up of Beverley in the early 2000s, despite improved prices of uranium and short-lived bursts of uranium exploration activities in the 1990s [9]. The Honeymoon mine produced a small amount of uranium from 2011 – 2013 but was placed on care and maintenance due to a combination of technical problems and the weakened uranium market [10]. Honeymoon has released a re-start strategy and is looking to resume production soon. As of December 2019, Australia has three operating uranium mines, namely Ranger, Olympic Dam and Beverley (mined from Beverley North and Four Mile deposits).

Ranger is an open-pit uranium mine located within (but not a part of) the Kakadu National Park in the Northern Territory and is currently owned by ERA. Discovered in the late 1960s, Ranger has been operating since 1980 and has produced over 130,000 tonnes of UOC to this day [11]. The deposit has, on average, a uranium grade in the order of 0.15–0.25% U_3O_8 [9]. As the mining of ore from their No. 3 pit was completed in 2012, ERA has been processing ore from stockpiles and will continue to decrease production as they are required to cease all processing activities by 2021 and complete rehabilitation works and final closure by 2026 [12].

Olympic Dam is an underground copper and uranium mine, located 560 km north of Adelaide in South Australia, and is managed and operated by BHP. It is the largest known uranium deposit in the world, containing an estimated total reserve of over 2,500,000 tonnes of uranium [9]. At ~0.05% U_3O_8, the deposit is considered to be low grade, however the presence of copper makes the operation viable (up to 75% of Olympic Dam's revenue comes from copper sales) [9 & 13]. Over the past decade, with production decreasing at Ranger, Olympic Dam has become Australia's largest producer and exporter of uranium. BHP has plans to augment the mines processing capacity for both copper and uranium within the next decade.

Beverley is an ISL mine and processing plant located approximately 520 km north of Adelaide, and is owned and operated by Heathgate Resources. Officially opened in 2001, the main Beverley deposit has been the primary source of uranium for Heathgate. Additional mining grants in 2008 and 2011 have enabled Heathgate to mine uranium using ISL from the Beverley North and Four Mile deposits, both of which contain around 0.3% U_3O_8. As the primary uranium processing plant is located at Beverley, these additional mines have been operated as satellite mines, whereby uranium extracted using ISL is loaded onto IX resins at satellite plants, with the loaded resin then trucked to the main processing plant at Beverley for stripping, precipitation and drying. In recent years, Beverley / Four Mile has contributed around 20% of Australia's total UOC exports [14].

IV. Challenges of Uranium Mining in Modern Australia

Of the 30 major Australian uranium deposits that have been discovered but not yet mined, six have approval to operate, and yet only three are operational [14]. These simple figures highlight how uranium mining in Australia is held back by several compounding factors including, but not limited to, economic, technological, environmental, social and political factors.

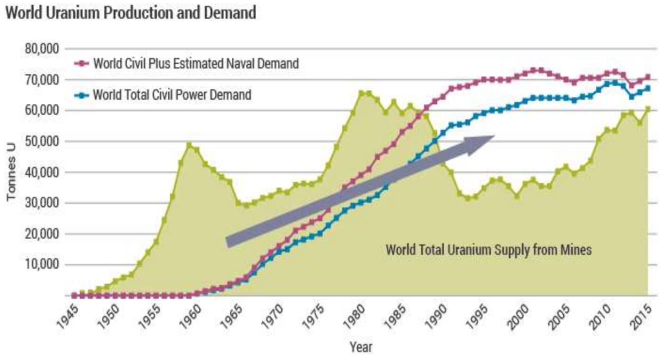

Figure 1. Global Uranium Production and Demand [15].

A. Economic and Technological Issues

The most dominant factor influencing the uranium industry in Australia is the price of uranium. From the all-time high spot price of almost $140 USD/lb in 2007, the price of uranium has dropped and remained at around $25 USD/lb since 2016. This is despite the global demand for uranium not being met from uranium production for the past few decades (see Fig. 1), and indicates that increases in reactor efficiencies and sourcing of uranium fuel from secondary sources have had a major impact on the viability of the uranium mining industry.

Accompanying the fall in the price of uranium, Australian deposits that were economically viable and profitable started recording losses [16], and approved mines that were undergoing development and even commissioning phases were put on hold. The impact this had on uranium mining in Australia was varied. Ranger mine, which was already phasing out production, recorded net losses in 2016 and 2017, and even resorted to purchasing UOC on the spot market (instead of producing the uranium at a higher net cost) to meet their sales requirements [16]. Further, ERA's Ranger 3 deeps project was placed on hold by the company's board until the conditions of the uranium market improved. With uranium as their sole commodity, the existing ISL operation at Beverley/ Four Mile suffered a decrease in revenue, but the superior cost effectiveness of ISL over open-pit/underground mining has allowed this operation to continue in production. Boss Resource's ISL mine at Honeymoon was commissioned in 2011, just after the initial drop in the price of uranium. However, Honeymoon's operations ceased in 2013 due to the fall in the uranium market coupled with technical issues. As previously stated, less than 25% of Olympic Dam's total income is from UOC production, so the impact was not as significant on BHP.

With conventional downstream processing of uranium so well understood, proven and optimised over several decades, the mining method used to extract the ore, the grade of the ore and the reagent consumption of extracting the uranium are the most dominant factors in determining the economic viability of a deposit. Open-pit and underground mines targeting low-grade deposits are becoming less common given the high operating expenditure required to mine, crush and in some cases beneficiate the ore. As such, unless there is another saleable commodity associated with the ore (e.g. Olympic Dam), these conventional mining methods are now trending towards being only used for higher grade deposits, for example at the McArthur River and Cigar Lake mines in Canada, which have the highest grade deposits in the world. While not all deposits can be mined using ISL mining, ISL mining processes, and to a lesser extent heap leaching, are now preferable for lower grade ores.

The economic shift has placed emphasis on the importance of technology in the uranium mining industry. Globally, as shown in Fig. 2, there has been an increased trend towards the cheaper but more technologically dependent ISL mining method for suitable deposits. Similarly, as Australia's open-pit mining at Ranger winds down production and the ISL operations at Four Mile approach full capacity, the percentage of ISL mining in Australia has increased proportionally, as shown in Fig. 3. Kazakhstan, the world's biggest producer of UOC, now mines all uranium using ISL methods since the closure of their open-pit and underground mines in 2013 [17]. Importantly, the majority of Kazakhstan's uranium resources are in sandstone deposits, which are amenable to ISL mining [18].

Excluding the deposit at Olympic Dam, Australian deposits amenable to ISL mining make up almost 30% of Australia's inferred uranium resources [14]. Technological constraints have been impacting deposits, such as Honeymoon, from becoming profitable in today's uranium market. One of the largest challenges Honeymoon has faced has been a result of the environment in which the uranium deposits reside. The deposits' aquifers possess elevated chloride concentrations, which significantly affect the ability for the uranium to be recovered by IX [19]. A series of test work programs undertaken by the Minerals team at ANSTO has demonstrated a new high performance IX process that significantly increases the recovery of uranium when operating at these elevated chloride levels [19].

In combination with an assessment of alternative flowsheet options to improve overall process efficiency, Boss Resources is now considering the restarting of operations after completing a Definitive Feasibility Study.

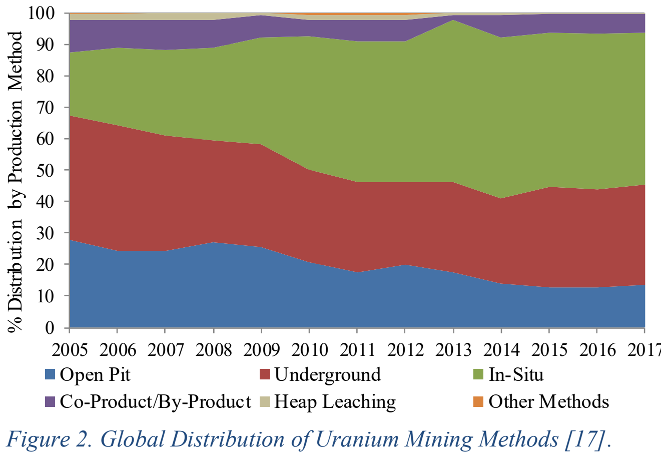

Figure 2. Global Distribution of Uranium Mining Methods [17].

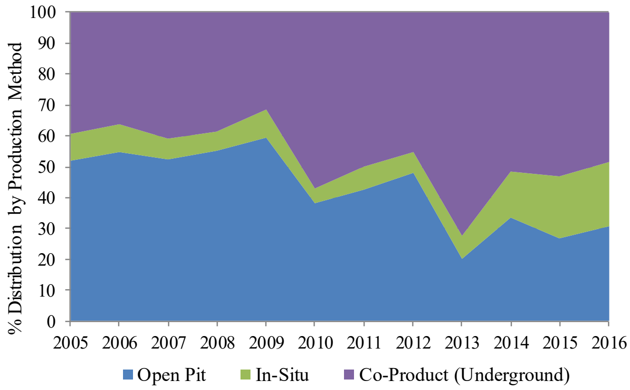

Figure 3. Australian Distribution of Uranium Mining Methods [20].

B. Social and Political Factors

Public perception and opinion is particularly important for the uranium mining industry, as several political movements over the past few decades have been against or in support of uranium mining in order to win voters [21]. According to recent poll results, shown in Fig. 4, public opinion regarding the export of uranium from Australia to other countries for peaceful purposes has remained relatively consistent over recent years [22]. However, the degree of uncertainty in the respondents has increased in recent years, so the trend is not conclusive and suggests a lack of awareness or knowledge of the uranium industry and Australia's participation in the nuclear non-proliferation treaty. Interestingly, support for the construction and operation of nuclear power plants in Australia as a source of 'cleaner energy' has increased from 35% in 2011 to 51% in 2019, although the uncertainty has similarly increased from 7 to 15% [22].

Where other Australian commodity mining activities require State Government approval only, uranium mining activities require both State and Federal Government approval [21]. The Minerals Council of Australia is advocating that this be changed as it adds a layer of duplication and delay [23]. The effect of this law was seen in the 1980s, when the Federal Labour Government brought in the '3 mine policy', restricting Australian uranium mining operations to Ranger, Olympic Dam and Nabarlek, as mentioned in Section III. While this policy was scrapped in 1996 by the Coalition and later given on-going support by the Federal Labour Government, other introduced State Government policies and laws restricting uranium mining have had a big impact and brought uncertainty to the industry. Uranium mining is currently only permitted in the Northern Territory and South Australia, and is banned in all other states and territories. Western Australia has recently banned uranium mining, although the Labour Government has agreed that mines approved under the former Liberal Government can proceed [24]. Similarly, the Queensland Labour Government has also recently banned uranium mining, however no mines had been approved by the previous Liberal Government.

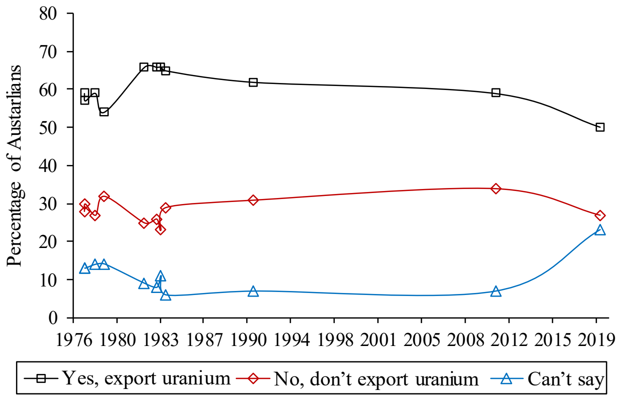

Figure 4. Australian Poll on Export of Uranium for Peaceful Purposes [22].

While several policies and laws in place currently restrict uranium mining and nuclear activities in Australia, recent developments show the potential for change in the industry. NSW has recently permitted uranium exploration activities and a Government inquiry is currently being held to determine whether uranium mining and other nuclear activities should be permitted in the State. Victoria is similarly holding an inquiry into uranium mining and nuclear related activities in the state. Further, the Federal Government is currently holding an inquiry regarding the broader use of nuclear energy in Australia.

V. CONCLUSION

Uranium mining and processing in Australia has a long history, with significant impacts from political, economic, technological and environmental factors. The current status of uranium mining in Australia is reflective of the uranium market, with several approved mines awaiting an upward turn in the price of uranium. Ranger is processing stockpiles and producing UOC in order to off-set shutdown costs, Olympic Dam continues to produce UOC as a result of their copper production, and Beverley continues production using more cost effective ISL mining techniques. The potential for new mines is ultimately contingent on economics as well as governmental policies, hence the current focus on reducing carbon emissions worldwide may provide opportunities for the uranium industry in Australia as well. Where economic barriers are the dominant force for unviable uranium mines, continued investment in research, process development and optimisation provide additional promise for future uranium production.

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Contributions To The Uranium Ore Processing Flowsheet By ANSTO's Minerals Business Unit

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I. INTRODUCTION

Uranium mining and processing is a mature industry. The basic flowsheet for processing uranium ore has not changed significantly in the last 60 years. However, each deposit has its own unique combination of uranium grade, mineralogy, geology, total resource and regional characteristics (climate, remoteness, water quality etc.). These, plus a drive towards cheaper and cleaner operations, means a critical review of each process is still warranted for both developing and established mines to ensure they are operating at their full potential.

ANSTO's Minerals Business Unit (ANSTO) is a research and consulting group comprised of more than 60 professional scientists and technicians with expertise that covers chemical engineering, metallurgy, mineralogy, chemistry, geology and radiation safety. This expertise is coupled with a multi-discipline facility, which has the capability to support an array of test work, ranging from bench scale analysis through to continuous demonstration plant operation.

ANSTO has developed a reputation as a world leading consultancy group in the development and testing of commercial uranium mining flowsheets. ANSTO has worked in the field of uranium processing for over 40 years, providing support to all of Australia's past and current operating sites including, but not limited to, Olympic Dam, Ranger, Beverley, Honeymoon, Yeelirrie, Kintyre and Four Mile. ANSTO has also worked on a wide range of overseas projects such as the Rössing and Langer Heinrich operations in Namibia, as well as other projects in Africa including Trekkopje, Mkuju River, Kayelekera and Falea.

Projects undertaken have covered all key aspects of the uranium flowsheet including: acid/alkali leaching, ion exchange (IX), solvent extraction (SX), resin-in-pulp (RIP), uranium product precipitation, reagent recovery and waste neutralization. The scale of this work has been quite diverse, including lab test programs, continuous pilot plant operations, on site pilot trials, and site process and operation reviews.

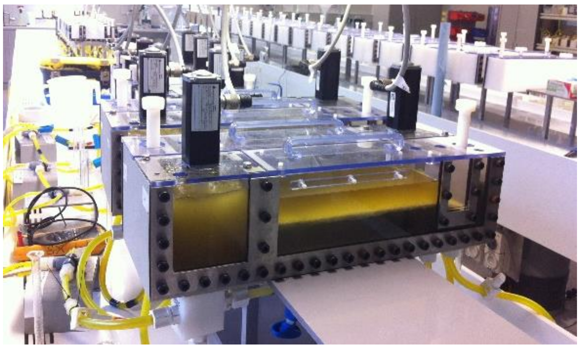

Figure 1. Solvent extraction circuit operated at ANSTO

II. PROCESS DEVELOPMENT AT ANSTO

The process development methodology applied by ANSTO primarily consists of four stages including:

- Mineralogy and characterization;
- Flowsheet development and related laboratory test work, including management of radioactivity;
- Techno-economic analysis;
- Continuous piloting and/or product generation.

Upon identification of a potential deposit, characterization is the first step in process development. Once the ore body has been discovered preliminary assays and mineralogical assessments are used to confirm the ore type, uranium mineralogy and grade. At this point challenges in processing an ore for uranium recovery can be first highlighted. Examples include the degree of liberation of uranium minerals, identification of refractory uranium bearing minerals within the deposit, the extent of reagent consuming gangue material, the presence of other problematic material (e.g. clays, organics) and the presence of other commercially significant by-products.

Ore body characterization is utilized to inform the selection of appropriate operating conditions and unit operations for flow sheet development. This may consist of industry established practices, a 'novel approach' incorporating new technologies developed by ANSTO, or a combination of both. Radioactivity management may also be assessed using desktop deportment studies. In many cases, the deportment of radioactive species is critical to the success of an operation, from both a product quality stand-point and work health and safety implications.

With a proposed flow sheet nominated, a techno-economic assessment can commence to define the economic viability of the project and range of options, e.g. SX or IX. The assessment typically encompasses both capital expenditure (CAPEX) (i.e. equipment selection), and operational expenditure (OPEX) associated with the flowsheet, or specific unit operations, subject to the client's requirements.

The final stage of process development is typically a demonstration of some, or all, of the unit operations of the flowsheet. This can take the form of either continuous or batch-wise operation. Data obtained and outcomes of test work can then be fed into engineering studies. The scale of the demonstration work is largely dependent on the client's requirements and the life cycle of the project. In the early development phase, or during the implementation of an operational improvement, a bench scale operation may be appropriate due to financial and/or time constraints of the project. Where a project is established or nearing execution, a more definitive demonstration may be necessary. This can be in the form of a pilot plant or demonstration plant operation, with multiple unit processes run continuously for extended periods of time.

Each stage of the process development methodology is intrinsically linked, with discoveries in one area typically informing a review of another. It is not uncommon for a process to be subjected to multiple reviews and further optimization over the life of the project.

The case studies presented in this paper represent examples where ANSTO was engaged as an expert consultant, by two uranium mining operations at different stages in their project life cycle. ANSTO's extensive experience across a variety of established and developing uranium operations, enabled the provision of timely and cost-effective process improvements in each scenario.

III. Case Study

A. *Uranium Recovery From Laterite at Ranger*

Ranger is an open-pit uranium mine located within (but not a part of) the Kakadu National Park in the Northern Territory. It has been operating since 1980 and will cease operation in 2021, with rehabilitation exercises concluding in 2026.

During the earlier periods of operation, the flowsheet involved the processing of a primary ore (~ 0.3% U_3O_8) via an acid leach, counter-current decantation (CCD) circuit for solid/liquid separation, SX, and through to ammonium diuranate (ADU) precipitation. Weathered material, known as laterite, within the ore body had been separately stockpiled due to a poor uranium extraction when leached with the primary ore [1]. Plant trials leaching the primary/laterite blend had resulted in extractions as low as 53%, compared to 90 – 92% when only processing the primary ore. The results suggested the poor extractions may have been attributed to a range of factors such as; the liquor composition in the leach, adsorption of uranium onto the solid, and/or the presence of refractory uranium mineral in the laterite ore. ANSTO was contracted by Ranger to assist in developing a process for recovering uranium from the laterite ores, as the reasonable grade of the material (between 0.2-0.5% U_3O_8) represented a valuable source of uranium.

The leach conditions were first investigated with attention to the solids content and pH – see **Table I**. Increased uranium extractions were observed at 45% solids when leaching between pH 1.0-1.5, compared to the primary ore leach conditions of pH 1.9. The trade off, however was a notable increase in acid consumption, which at pH 1.0 was up to five times that for the primary ore leach conditions. However, at a 2% solids pulp density, extractions from laterite were favorable, with results significantly better than the equivalent 45% solids leach at pH 1.9. This indicated that poor extractions may have been due to soluble uranium adsorbing onto solids in the leach (preg-robbing) and not a result of the presence of a refractory uranium mineral. To confirm this, resin-in-leach (RIL) was trialed using a range of ore grades and a strong base resin – see **Table II.** The resin competed with the preg-robbing process, with increased extractions across all tests containing IX resin. Lastly, particle size was assessed to investigate what proportion of the laterite was responsible for the preg-robbing and therefore what fraction could be processed via the primary leach (at pH 1.9), and in turn minimize the overall acid usage - see **Table III.** The leach performance of the +600 µm fraction under primary leaching conditions was deemed acceptable with 64.0% uranium extraction (based on liquors). The results also demonstrated that the challenges lay with the finer material, with finer fractions requiring more aggressive conditions to be leached effectively.

The finding of ANSTO's test work was the occurrence of preg-robbing when the lateritic ore was subjected to the primary leach conditions. Given the Ranger plant was already operational, and the introduction of a RIP circuit would be problematic, a cost-effective improvement was the introduction of a two-stage leach process. The laterite was screened prior to leaching, with the coarse > 1 mm fraction directed to the primary leach (pH 1.9 and 50% solids), while the remaining fine fraction was subjected to a more aggressive acid leach at pH 1.5 and 30% solids. The two slurries were then combined to produce a 7% laterite feed to the CCD stage. The plant trial of this configuration culminated in an overall uranium extraction of 83% in the leach circuits [2].

TABLE I. EXTRACTION OF U FROM LATERITE ORES

Laterite Ore Type	Leach pH	H_2SO_4 Consumption (kg/t)	U Extraction (%)	
			45% Solids	*2% Solids*
A	1.9	19	74	89
	1.5	36	85	-
	1.0	103	90	-

Laterite Ore Type	Leach pH	H_2SO_4 Consumption (kg/t)	U Extraction (%)	
			45% Solids	*2% Solids*
B	1.9	12	70	79
	1.5	23	78	-
	1.0	53	81	-

TABLE II. RIL LEACH OF LATERITE ORES

Laterite Ore Type	Grade (% U_3O_8)	IX	Leach pH	U Extraction (%)
A	0.456	No	1.9	73.8
		Yes	1.9	84.0
B	0.178	No	1.9	69.5
		Yes	1.9	71.2
C	0.314	No	1.95	48.4
		Yes #	1.9	62.1
# 2.3 vol.% resin used for this test. All other IX tests used 5 vol.% resin				

TABLE III. LEACH OF LATERITE ORES AT DIFFERENT SIZE FRACTIONS

Fraction (µm)	Leach pH	Grade (% U_3O_8)	U Extraction (%)	
			Solids Based	*Solution Based*
-1000 +600	1.95	0.308	85.0	64.0
-600	1.5	0.559	84.4	85.4
-1000 +150	1.95	0.400	80.3	37.0
-150	1.5	0.616	80.0	77.8
-1000 +75	1.95	0.402	72.6	32.2
-75	1.5	0.605	79.2	87.1

Figure 2. Resin-In-Pulp circuit used on-site by ANSTO

B. Mkuju River Process Improvement

Mkuju River is a uranium project located in Southern Tanzania owned by Uranium One (project was developed by Mantra Resources). As of 2013 the project had measured and indicated resources of 48,000 tU plus inferred resources of 10,600 tU at an average grade of 0.026% U. In 2017, the project was suspended due to the low uranium market prices [3].

In 2009, ANSTO was engaged by Mantra Resources to review the proposed flow sheet and investigate improvements to both OPEX and CAPEX as part of a definitive feasibility study (DFS) for the project. The review identified that a fairly intensive leaching step had been proposed. This included a two-stage leach, addition of ferric ion and oxidant, a CCD circuit, and a high acid addition in the leach. These operations were typical of controls needed for uranium extraction from clay/carbonaceous material (i.e. laterite) and to deal with issues arising from their processing, such as preg-robbing and poor settling characteristics.

Through the application of a diagnostic leach test program, ANSTO was able to quickly confirm the occurrence of preg-robbing and its contribution to the limited uranium extractions. Using the knowledge obtained in previous investigations (see above for Ranger laterite project), a RIP stage was developed and piloted at ANSTO. The results of this work were then applied to an engineering and metallurgical trade off study between the RIP and CCD processes [4].

The outcomes of the test work and trade off study were:

- Increased uranium extractions in the leach stage between 87-90 %, without the need for ferric ion and oxidant addition;
- The removal of the CCD circuit and subsequent simplification of operation;
- A reduction in reagent use when compared to the costs of a RIP process;
- Process improvements were identified in a timely manner and captured within the DFS

Partnering with the client for the process review enabled investigation of novel approaches and optimization of the flowsheet to overcome the challenges presented. In doing so significant improvements to the OPEX and CAPEX of the project were recommended and able to be trialed.

IV. CONCLUSION

ANSTO's engagement by Ranger to assist in developing a process for recovering uranium from laterite ores, and by Mantra Resources for the Mkuju River DFS review, are examples of where critical technical reviews of processes can have significant beneficial impacts on process outcomes.

In the Ranger case study, an investigation into the leach conditions for laterite ores found that improved uranium extractions could be achieved when the leach pH was reduced to 1.0 (i.e. greater acid content), or when the solids content was reduced to 2% solids. These results supported the proposition of soluble uranium adsorbing onto the solids within the leach, known as preg-robbing. This was confirmed through a RIL trial in which increased uranium extractions were observed across all tests containing IX resin. As the project's operational constraints meant that RIL could not be implemented, a further investigation into particle sizing showed that lateritic feed greater than 600 µm could be processed via the standard leach conditions, with only the finer fraction requiring different conditions. The culmination of ANSTO's work was the implementation of a two-stage leach, consisting of the existing Ranger leach conditions processing the bulk of the feed, and a second more aggressive leach to process the <1 mm fraction of the feed.

With the Mkuju River DFS review, the occurrence of preg-robbing was swiftly identified by ANSTO using the methodology and learnings acquired in part from the Ranger project. The status of the project meant the use of RIP was a viable solution, and in turn was efficiently developed and trialed within the time constraints of the DFS. The outcome was a simplified process with the removal of unnecessary unit operations and reduction in reagent requirements.

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Increase of energy potential of the natural uranium for nuclear power plants by using fast reactors

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I. Introduction

World nuclear energy in a sustainable development concept should follow the next requirements:

- safety guaranteeing of nuclear facilities in all operational modes;
- nuclear non-proliferation;
- economic competitiveness in process of nuclear energy production;
- limited influence on environment and compensation of ecological harm whenever possible;
- unlimited source of fuel in the nearest future.

Lack of solution for any of mentioned points is considered to be a barrier for world nuclear energy evolution.

Safety is an indisputable requirement of existence and possible development of nuclear energy.

Together with nuclear non-proliferation, the safety is implemented by constant improvement of systems, elements, and management methods, considering achieved level of operating experience, art and knowledge of nuclear energy facilities.

Economic competitiveness is a universal characteristic of industry as a functioning system.

Limiting the influence on environment is realized by improving the design, technological processes and proper waste management.

Uranium and its derivatives from nuclear reactions are considered the main fuel for nuclear energy worldwide today.

Talking about the requirements, the first four could be managed by human, but the fuel potential nowadays is limited by the natural uranium reserves. Without a strategy of fuel potential increase, the long-term perspectives of the industry development are indeterminate.

Implementing new fuel cycles, like thorium cycle, for example, could be considered as one of possible ways to increase the fuel potential. The thorium natural reserves are more than uranium reserves, but they are limited too.

In this article it is reviewed the complex approach of increasing the fuel potential and attendant decreasing of radioactive waste influence on environment by implementing the fast reactors into nuclear energy and by usage of accumulated depleted uranium and uranium irradiation products.

II. The State of Modern Nuclear Energy

Due to IAEA information in 2018 worldwide were operated 451 power plants with installed capacity of ~ 400 GW (e) and there were 55 power units with project installed capacity of ~53 GW (e) in building process [1]. From the mentioned power units only 3 have fast reactors: BN-600 & BN-800 (Russia); CEFR (China). Only in Russia the fast reactors are operated by the industry.

Modern state of the world nuclear energy can be described as based on operating thermal-neutron reactors and long storing of spent fuel.

On average fuel enrichment for thermal-neutron reactors by ^{235}U isotope represent 4-5%. For the generation of 1 GW (e) 170 tons of natural uranium are required per year [2], from this quantity 24 tons of fuel would be produced. After irradiation in open fuel cycle there will be produced about 24 tons of spent fuel and 146 tons of depleted uranium from 170 tons of natural uranium.

Some countries implement regeneration of MOX fuel and repeatedly one-time irradiation of produced fuel in a thermal-neutron reactor. Those actions increase fuel potential, but do not resolve the lack of resources in a long-term perspective. While repeated fuel irradiation in thermal-neutron reactor, even-numbered isotopes and minor actinides, which absorb the neutrons, are accumulated. Such spent fuel must be disposed, because next reprocessing and further irradiation in thermal-neutron reactor are getting impossible or economically unprofitable with a current state of art and knowledge. Such spent fuel will have to be disposed.

Elements having a slight half-life period (more than 25 years) provide the main gamma-activity and heat release of spent fuel after some years of maintaining. During the longer maintaining period (more than 100 years) transuranic elements having long half-life period as neptunium, americium, curium (minor actinides), produced by the nuclear reactions, present the major hazard. It makes difficult to design, build and control the

Rosatom State Corporation

repositories of spent nuclear fuel in long-term perspective due to all those circumstances.

The decay heat drops rapidly in about first 200 years. Later the decay heat is caused almost by the actinides and plutonium. This lasts up to 10,000 years. The relatively long half-life of the isotopes is the main reason of slow decrease of the decay heat [3].

III. Role of Fast Reactors

The use of fast reactors after reprocessing instead of reuse in thermal reactor bring several technical advantages with the objective of nuclear waste volume and activity reduction.

In comparison with thermal-neutron reactor the fast reactor has the following key advantages:

- fast neutrons fission wider variety of isotopes compare to thermal neutrons;
- quantity of secondary neutrons appeared while fission the main isotopes is more for high energy neutrons than for thermal neutrons.

The necessity to provide higher nuclear concentrations of fissile isotopes owing to low fission ratio of nuclear reactions for high energy neutrons becomes one of disadvantages of fast reactor. This gap is compensated by increase of fuel enrichment.

The advantages of fast reactor allow realizing inside the core some processes related to capture of surplus neutrons without negative influence to physics and economical operation of reactor. Among such processes are reproduction of secondary fission isotopes and transmutation of long-lived isotopes, i.e. transforming them to fission products. This improves radioecology of fuel cycle. Reproduction of secondary fission isotopes allows increasing the share of ^{239}Pu fissile by any neutrons and decreasing the share of isotopes that capture neutrons in regenerated fuel.

Basing on that characteristic it's possible to increase significantly the fuel potential of thermal neutron reactors by operating uranium-plutonium fuel, extracted from spent fuel of fast reactor. And decrease the quantity of accumulated spent fuel of thermal-neutron reactors.

Using regenerated uranium-plutonium fuel allows realizing the strategy of closed nuclear fuel cycle.

To realize the strategy of closed fuel cycle and long-term strategy of radiation equity of disposal it's necessary to invent the production of reprocessing and fractioning of spent fuel with multipurpose extraction as well.

While extracting the fractions of the fast reactor spent fuel according to their half-life period and physical-chemical characteristics it's possible to realize the following process:

- extraction of short-lived isotopes of fission products; controlled disposal of them during 100-200 years by decreasing activity to acceptable state for final disposal or for usage as radioisotope products for civil use;
- extraction of minor actinides to a separated fractions, or homogeneous production of uranium-plutonium fuel with minor actinides.

Technological process of closed fuel cycle may be represented as following:

- processing of uranium-plutonium fuel from the spent fuel of thermal-neutron or fast reactor;
- irradiation of fuel in core of fast reactor and "burning" of minor actinides (within fuel or in separated irradiation unit);
- reprocessing and fractioning on a regeneration plant the spent fuel extracting the fissile isotopes;
- processing of uranium-plutonium fuel for thermal-neutron or fast reactor.

Applying the technologies allowing maintaining the homogeneous reprocessing of uranium-plutonium fuel without extraction of exact fissile isotope will influence positively on a nuclear non-proliferation regime. Capacities of extracting the minor actinides from the spent fuel are generated now.

Technological process of closed fuel cycle doesn't lead to generation of depleted uranium; it decreases quantity of waste disposal in comparison with open fuel cycle.

As mentioned above, while processing the reactor with installed capacity 1 GW(e) there is produced 24 tons of spent fuel (see Chapter II: The state of modern nuclear enegry). This fuel consists from U, Pu, fission products and long-lived isotopes. Uranium and Plutonium could be returned to the fuel cycle. Fission products and long-live isotopes present approximately 5% of total. Thereby approximately 1.2 tons of fission products and long-live isotopes have to be disposed finally.

After implementing the technology of fractioning and differential repository according to half-life period and physical-chemical characteristics of the fractions, the quantity of disposed elements will significantly decrease, as well as their radioactivity. This will decrease expenses for repository of nuclear industry waste and will increase the safety of repository in a long-term period a lot. Prospectively, it is necessary to get closer to radiological equity of extracted uranium and of the waste from closed fuel cycle.

Fast reactors used together with reprocessing and fractioning allow achieving the radiological equity of disposal. The radiological equity approach at the closed fuel cycle is the main way to solve potential environmental problems when dealing with radioactive waste. It actually means that the environmental radiation safety is secured by the very absence of radioactivity above the already existing natural levels.

Important to note, that it is essential for the industry to solve the points related to processes automatization to provide the radiation protection of stuff.

IV. ROSATOM ACTIVITY ON THE FAST REACTORS FIELD

Rosatom is researching the development of the technology of the closed fuel cycle now. Tests and researches of uranium-plutonium fuel based on BN-600 and BN-800 reactors of Beloyarsk NPP are held.

Reactor tests of mixed nitride uranium-plutonium (MNUP) fuel were held in the BN-600 reactor. There were performed post-irradiation examinations of nuclear fuel elements with various burnup. It confirmed the efficiency of technologies used for fuel fabrication and the manufacture of fuel elements and fuel assemblies with MNUP fuel. The first fuel elements with MNUP fuel and minor actinides have been manufactured and supplied for irradiation [4]. The questions of influence of the initial content of MNUP fuel and the impact of the initial parameters to the possibility of fuel self-sufficient mode of the reactor during the whole period of its operation are studied. Rosatom conducts reactor tests and post-irradiation examinations with MNUP fuel to examine the opportunity to use it in BREST-OD-300 and BN-1200 reactors [5].

Serial production of MOX fuel started in 2018. The world's most powerful fast neutron reactor BN-800 has been loaded with the first batch of MOX fuel made of depleted uranium and plutonium oxides. The power unit has successfully resumed operation in 2020 after an overhaul. The core with a full load of MOX fuel would be finished gradually in next two years [6].

V. CONCLUSION

Implementation of fast reactors to nuclear industry will increase fuel potential more than 100 times due to use of accumulated depleted ^{238}U.

Migration from the modern state of world nuclear energy, based on operating the thermal-neutron reactors and long repository of spent fuel, to the closed fuel cycle based on fast reactors with reprocessing of accumulated and currently being generated spent fuel, would make the world fuel potential almost unlimited.

Development and implementation of transmutation technology of long-lived nuclides and fractioning of spent fuel, allow to decrease the influence of nuclear industry on environment to an acceptable level and to get closer to a radioactive equity.

Spent fuel storage cost reduction may be achieved by diminishing the need of building large repositories, thanks to the use of the accumulated depleted uranium. This in turn would benefit the economy of the nuclear industry. Today the industry needs to develop new technologies at all stages of the life cycle to improve its economic efficiency.

Development of fast reactor technologies and reprocessing technologies are complex solution to realize a concept of sustainable development of world nuclear energy.

Rosatom is implementing the strategy of dual-component nuclear power system with both thermal neutron and fast neutron reactors and establishing the technological base for close nuclear fuel cycle with fast neutron reactors as its main component.

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Radioactive Waste Repatriation from the United Kingdom

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I. Introduction

In 1996, the Australian Nuclear Science and Technology Organisation (ANSTO) transported spent nuclear fuel to the UK for reprocessing. The fuel sent to the Dounreay facility in Scotland has since been reprocessed, with the waste residue that was generated incorporated into a cemented form. As part of an international understanding between the Australian and UK governments, ANSTO is obliged to repatriate an equivalent quantity (in terms of activity) of radioactive waste, and will do so in vitrified form.

ANSTO is therefore coordinating the evaluation, transport and storage of the waste in a major undertaking that requires the collaborative efforts of numerous international stakeholders. By managing this radioactive waste safely and securely, ANSTO will have fulfilled its international obligations in the nuclear fuel cycle.

II. Project

A. Overview

For a project of this complexity, the return logistics and the lengthy national and international regulatory approval processes have meant detailed planning, stakeholder engagement and cost estimation. One of the major deliverables of the project was the construction of a specially designed 100 tonne cask in which the waste canisters can be transported. The cask was manufactured in Belgium and fully validated, with testing of equipment from overseas to ensure that all the components were fully compatible in their fully assembled configurations.

In line with the international schedules for nuclear shipments, the empty cask will make its way to the Sellafield nuclear site in the UK where the radioactive waste designated for repatriation to Australia is currently stored. There, the waste material will be evaluated for its radioactive content and loaded into the cask. The next phase will then see the laden package shipped from the UK to Australia (see Figure 1). The project has already overseen the construction of an interim storage facility at Lucas Heights where similar waste from France is now stored and where the UK waste will be stored upon delivery.

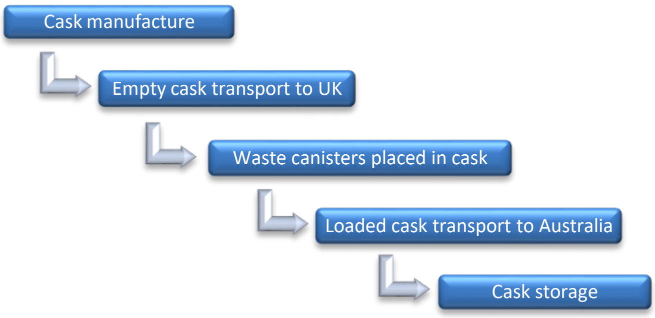

Figure 1. Project milestone stages

B. Conditioning of waste

Processes such as cementation and vitrification are used to convert waste into a stable solid form that is both insoluble and non-dispersible. Cementation involves the immobilisation of waste through the use of specially formulated grouts mixed in with radioactive sludges and flocks and allowed to set as a solid volume of concreted waste. Vitrification involves the incorporation of waste into a glass matrix. Initially, the residues are calcined to a granular powder before being combined with molten glass and poured into a robust stainless steel canister. The mixture is then allowed to cool, forming a solid matrix within the canister, which in turn is welded shut [1] (see Figure 2).

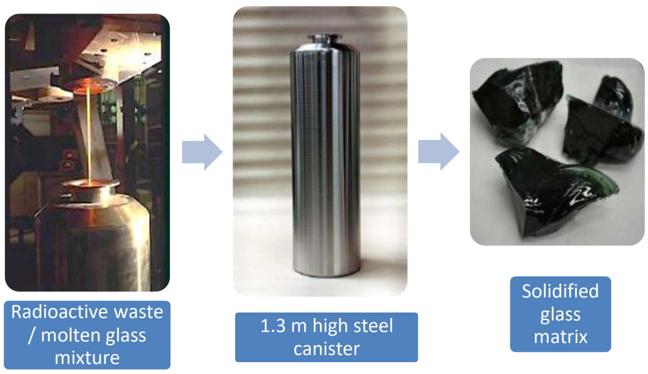

Figure 2. Vitrification process

Initially, ANSTO owned title to the cemented waste generated from the reprocessing of spent nuclear fuel at Dounreay. In 2010, the UK Government and Scottish Executive conducted a public consultation process to allow the substitution of cemented wastes by vitrified wastes. The Integrated Toxic Potential methodology was approved by both the UK Environment Agency and the Scottish Environmental Protection Agency to accurately determine waste equivalence between the two different processes.

Australia was therefore given the option of substituting the cemented waste with an equivalent quantity of vitrified waste instead. Given that vitrified waste occupies less than 5 per cent of the volume of an equivalent amount (in activity terms) of cemented waste, such a substitution offered significant cost benefits due to the reduced transport and storage requirements.

Given these advantages, ANSTO contracted with International Nuclear Services (INS), a wholly owned subsidiary of the UK Nuclear Decommissioning Authority, responsible for managing the return of vitrified waste for Dounreay customers. Under this contract, Australia transferred ownership of its cemented waste and took ownership of a radiologically equivalent, but far smaller volume of vitrified waste located at the Sellafield nuclear site.

C. *Regulatory Affairs*

Successful repatriation of the radioactive wastes requires that approvals be obtained from relevant regulatory bodies throughout Europe and Australia. In particular, the Australian nuclear safety regulator, the Australian Radiation Protection and Nuclear Safety Agency (ARPANSA), needs to be satisfied of a range of issues before the return is able to proceed. ANSTO is required to show evidence of matters such as:

- Licensing of the Interim Waste Storage (IWS) facility;
- Validation of the cask, including Safety Analysis Reports;
- Transportation details; and
- Parameters of both the original spent nuclear fuel and the vitrified waste designated for repatriation.

In addition to addressing ARPANSA's requirements, ANSTO will be required to either seek approval from, or inform, the following bodies:

- The Australian Government's Department of Environment and Energy, with respect to the Environment Protection and Biodiversity Conservation Act 1999;
- Australian Maritime Safety Authority;
- NSW Roads and Maritime Services;
- NSW Environmental Protection Authority; and
- Local councils and port authorities.

III. Repatriation

A. *TN 81 cask*

Figure 3. Drop test of 1:3 scale model of TN 81 cask at 5° incline

To manufacture a suitable cask for the vitrified waste, ANSTO decided to use the TN 81 Approved Transport package from Orano TN. The design is in line with international best practice, offering 50 years of storage with the possibility to extend the duration. It underwent rigorous tests to ensure satisfactory performance in radioactive shielding, heat dissipation and structural integrity (see Figure 3), and is the same as the cask which was used to repatriate the waste from France.

The TN 81 cask is designed for the transport and intermediate storage of up to 28 canisters containing vitrified waste. When fully loaded, the maximum allowable mass of the cask is 116 tonnes. The cask takes the form of a cylinder with overall dimensions 7215 mm long x 2750 mm diameter. The main components of the package comprise the following (see Figure 4):

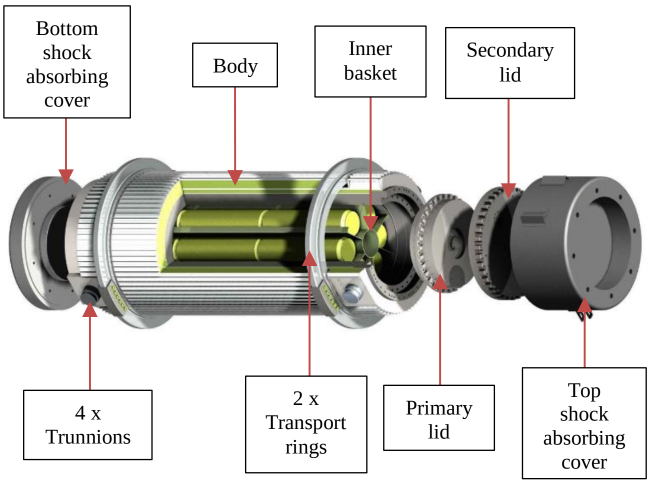

Figure 4. TN 81 cask

Figure 5. TN 81 cask in transport frame lifted by 4-legged lifting beam (inset right: 2-legged lifting beam)

- Body with a 200 mm thick forged steel shell, and outer aluminium plating providing gamma shielding, neutron shielding and heat dissipation features;
- Two pairs of trunnions for securing and handling of the cask – horizontally by four trunnions and vertically by two trunnions;
- Enclosure system which includes primary and secondary lids for use depending on the loaded content;
- Shock absorbing system which includes protective transport rings and detachable shock absorbing covers containing a wood-steel composite; and
- Inner copper basket consisting of seven housings on four levels designed to hold up to 28 canisters of vitrified waste.

In addition to third party certification of the cask by an independent inspector, it was necessary for ANSTO to ensure that the cask and all of its ancillary components were compatible with each other. This was an important risk management strategy – in anticipation of future transport and handling operations – requiring equipment to be shipped from Australia, the UK and Switzerland to the Belgian facility for complete interface testing. Such equipment included (see Figure 5):

- INS Transport Frame (for transport to and from the UK);
- ANSTO Transport Frame (for subsequent transport in Australia);
- Shock absorbing covers;
- Transport rings; and
- Two- and four-legged lifting beams.

The four-legged lifting beams engage with the four trunnions to manoeuvre the cask while in a horizontal orientation, whereas the two-legged lifting beams engage with the two head trunnions to manoeuvre the cask to and from a vertical orientation.

B. Loading and Transport

Figure 6. Class INF2 vessel

A set of parameters was used to evaluate the content of the vitrified waste at Sellafield. These included the isotopic content, thermal output and radioactivity levels, calculated with reference to the parameters of the spent nuclear fuel which had been sent to the UK in 1996. Inspections of the steel canisters containing the waste were also undertaken to ensure that they were of sufficient structural integrity.

In accordance with the international schedules for radioactive waste shipments, ANSTO must transport the vitrified waste within a set window following the loading of canisters in the shipment cask. Careful planning and logistics are required during transportation, including when transferring the TN 81 package to and from rail / road / sea modes of transport. Detailed transport plans therefore need to be in place well ahead of implementation.

Once the loading of the canisters has been completed and the cask is ready for shipment, the package will be transported to a designated UK port for loading onto a Class INF 2 ship (see Figure 6). This vessel class is certified by the International Maritime Organisation to carry radioactive waste [2]. The international logistics and scheduling for radioactive shipments of this nature are so regimented that the designated vessel will require booking a year in advance. For security purposes, efforts will be made to make the schedules and routes unpredictable.

C. Storage

ANSTO's strategy for storing repatriated waste involves interim storage at Lucas Heights until the National Radioactive Waste Management Facility (NRWMF) is completed. With the NRWMF not yet ready, ANSTO was required to construct an interim waste storage (IWS) facility (see Figure 7) at Lucas Heights in 2015 as part of ANSTO's integrated Waste Management infrastructure. The facility's initial inventory consisted of the reprocessed waste from France, which was also in vitrified form.

The quality of radioactive and thermal shielding in the TN 81 cask design is such that there was no need for additional shielding in the facility. The IWS is engineered to be resistant to foreseeable incidents such as earthquakes and severe weather conditions. It also provides protection from environmental threats such as atmospheric salts and high humidity levels, which could have an impact on the long-term integrity of the package.

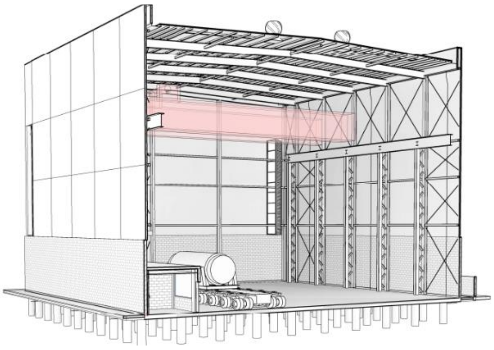

Figure 7. Interim Waste Storage facility

Therefore, to protect against an accidental radiological release of ANSTO's repatriated waste, the following engineered barriers are in place:

- the waste form – vitrified;
- the container – stainless steel canister welded shut;
- shielding – TN 81 design; and
- the external store structure.

The IWS was fitted with all necessary electrical and mechanical service systems, such as power, lighting, natural ventilation, fire detection, lifting equipment, security systems, etc. Also installed was a 140 tonne dangerous goods rated (170 tonne total rating) crane designed to unload the TN 81 cask from the transport truck and place it at a desired location inside the store.

The NRWMF is currently under development for long-term storage of Australia's radioactive waste. Upon establishment of this facility, the waste held at the IWS at ANSTO will be transferred to the NRWMF. The management of the waste at this long-term facility will conclude ANSTO's international obligations in the nuclear fuel cycle.

IV. Conclusion

Waste management is an integral facet of the nuclear fuel cycle. Having transported spent nuclear fuel to the UK in 1996 for reprocessing, ANSTO is now overseeing the repatriation of a dedicated quantity of radioactive waste. In order to manage this project effectively, ANSTO has engaged the services of various stakeholders, both internal and external, to provide expertise in scientific, engineering, logistical and regulatory areas.

Consequently, the entire repatriation undertaking has been a lengthy process, requiring the completion of milestones in successive stages. The remaining objectives will be finalised over the course of the next few years. This has been made easier by the fact that substantial groundwork has been laid to ensure their successful execution and that critical deliverables have already been successfully met.

Acknowledgment

The authors would like to thank ANSTO Waste Management Services for their contribution to this project.

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Application of Best Available Technique assessment on ANSTO legacy waste

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I. Abstract

The Australian Nuclear Science and Technology Organisation (ANSTO) has played a crucial role in allowing Australians to be able to benefit from nuclear medicine and research activities. Around 1 in 2 Australians will use nuclear medicine throughout their lifetime. This benefit presents the responsibility for using international best practice to identify the best methods to safely manage radioactive waste. Some of Australia's waste comes from the High Flux Research Reactor (HIFAR) which operated for around 50 years until it was replaced by the Open-pool Australian Lightwater Reactor (OPAL) in 2007. HIFAR supplied millions of doses of nuclear medicine, the production of which resulted in some legacy Intermediate Level Liquid and Solid Waste (ILLW & ILSW respectively) currently stored at ANSTO. In determining how best to condition that waste, ANSTO elected to use the Best Available Technique (BAT) process as this has been identified by the International Commission on Radiological Protection (ICRP) and the Australian Radiation Protection and Nuclear Safety Agency (ARPANSA) as representing international best practice. This paper details the implementation of the methodology and highlights learnings from its application. The technique allows a better-informed decision-making process and objective evaluation of the options available to process and condition the waste for final disposal/storage.

II. Introduction and Background

Over a 36-year period (1971-2007), ANSTO, previously the Australian Atomic Energy Commission (AAEC) produced Mo-99 through the acidic dissolution of uranium material from cans irradiated in HIFAR (Fig 1.). Acidic intermediate-level liquid waste from this process was transferred for interim storage. Approximately half of this waste was solidified in the past to reduce waste volumes, with the remaining waste still stored in tanks in liquid form. ANSTO is currently building the SyMo Facility to treat and manage the alkaline ILLW generated from its current Mo-99 production process through an immobilization process that applies synthetic-rock (SYNROC) technology. An alternative processing plant will be required for the acidic legacy wastes, as the SyMo facility will not be available for treatment of wastes other than the target Mo-99 alkaline wastes until the end of its operational life. This decision guided the opportunity for ANSTO to use the BAT assessment to evaluate options before proceeding with the research and development of a process that will produce a wasteform suitable for long term storage and ultimate disposal.

Figure 1. ANSTO's High Flux Research Reactor (HIFAR)

III. Applying the BAT Methodology

The BAT is an internationally preferred method to be employed for the management and disposal of radioactive waste. The assessment process was undertaken in accordance with the 'Nuclear Industry Code of Practice- Best Available Techniques for the Management of the Generation and Disposal of Radioactive Wastes' [1]. The general sequence used to conduct a BAT assessment is provided in Fig. 2.

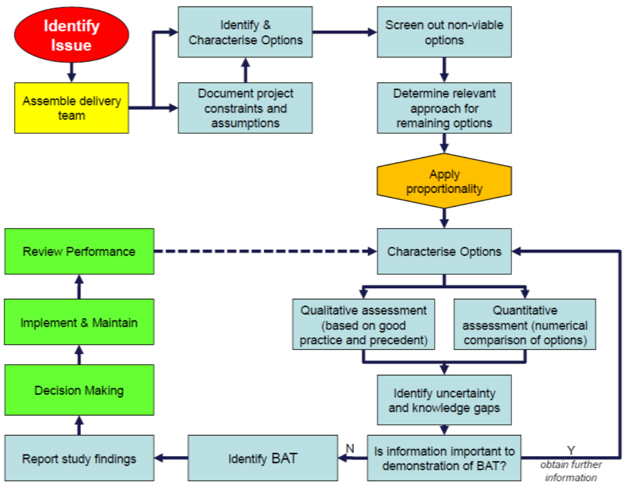

Figure 2. Best Available Technique (BAT) process [1]

In this process, the best solution is selected from a number of competing techniques after taking into account a number of factors such as radiation protection, economics, process technology, government policy and legislation. In particular, the following were considered:

- The scale of the challenge and scope of the project
- Established good practice, i.e. comparable processes, facilities or methods of operation
- Economic viability of the waste management technique
- Technological advances and changes in scientific knowledge
- Business strategy and regulatory consideration

Once the issue requiring a decision is identified, a delivery team is assembled. It is crucial that the delivery team is different to the team that participates in the decision-making process. This eliminates bias in decision-making and allows an organisation to decide on a waste management technique that objectively aligns best with their key drivers including but not limited to strategy, risk, timelines and budget required to deliver the option. When assembling the delivery team, it is important to identify the roles and perspectives needed. This team will be responsible for documenting assumptions and constraints, developing the scoring criteria and scoring the characterised options to identify the best available technique. For ANSTO's legacy waste, the team comprised experts across the organisation including government affairs, research and separation techniques, process engineering, SYNROC technology and waste operations. A project engineer and manager were responsible for leading and guiding the team through the process, as well as gathering the necessary information to allow the team to progress through each stage of the process.

As seen in Fig 2, once the options are characterised, they are narrowed down using a screening process, and the viable options to evaluate in more detail are identified. Work is then commenced to collect the relevant information that will allow the delivery team to assess the viable options and rank them. The delivery team's varying expertise allowed the successful scoring of the options against both qualitative and quantitative criterions. This process captures the assumptions in place when capturing the information that is scored. Knowledge gaps and any risks are also identified for the proposed options to ease planning of the option which progresses. Using these assumptions, timelines and class 5 estimate costs [2] are captured to assist in the scoring.

The front end of the BAT process is essential for laying the foundations for the stages afterwards through to the final decision. Once the correct roles and individuals are identified, the assumptions and constraints identified, and initial options screened out, the scoring criteria is developed. This allows the qualitative and quantitative assessment of the remaining options that follows. There are few things to consider when developing the criteria. These include:

*1) **Choose criteria that will differentiate options**: If options score the same for a criterion at the end, the criterion becomes redundant in aiding the decision making process. Choosing the best available technique needs criteria which allow the differentiators between options to be clearly identified.*

*2) **Minimise the number of criteria**: This ensures criteria which overlap are included together. When identifying each criterion to assess options, carefully select them to ensure they are aligned with the organisation's key drivers in a decision making process.*

*3) **Adapt a 'universal' scoring scheme**: The scheme used to allow the scoring of each criterion must be applicable to varying options. The wording and characterisation behind a scheme is essential as it will determine how an option will rate for each criterion.*

The scheme and definitions developed to score the ANSTO legacy waste included scoring each option as low, medium or high for each criterion. The delivery team used these scoring tables which clearly defined what constituted a low, medium or high score for each individual criterion.

The qualitative and quantitative scoring developed by the ANSTO delivery team comprised of ten evaluation criteria. Table 1 below provides a summary of the three quantitative criteria used for evaluations. Table 2 summarises the qualitative criteria used. The tables are an indication of the information gathered to allow the high-level assessment of the options. This approach at a pre-conceptual stage allows ANSTO to demonstrate its due diligence in dealing with radioactive waste through a robust process of decision making. This is crucial when dealing with wastes with higher doses and activity concentrations.

TABLE I. QUANTITATIVE CRITERIA

Criterion	Description
Waste product	Total waste volume, waste package size and type, dose rate (contact and at 1m) and activity mass concentration.
Time frame	Estimated timeline to deliver the option including research and development stages to final facility operation and waste processing
Cost	End to end capital and operational expenses (Research and development to disposal).

TABLE II. QUALITATIVE CRITERIA

Criterion	Description
Wasteform	Assessment against mechanical strength, resistance to impact, fire resistance, voidage, resistance to leaching, radiation stability, criticality issues, radiological safety and ease of accountability
Waste management suitability	Assessment against standards, including internal radiation, nuclear safety and waste management standards. The option is also assessed against the NRWMF generic waste acceptance criteria (WAC).
Regulatory compliance	Compliance including but not limited to Australian Radiation Protection and Nuclear Safety Agency (ARPANSA), Department of Energy and Environment (DoEE), Australian Safeguards and Non-proliferation Office (ASNO) and Comcare regulations.
Technology	Technology maturity and expertise required
Robustness	Complexity of start-up, operation and shut down as well as packaging and shielding requirements for storage and transport
Opportunities and future applications	Assessment of the option against other ANSTO waste streams and on possibilities for future applications of the proposed facility.
Risk	Political, economic, timeline and financial risks

Using the information gathered, the delivery team gather together in meetings to assess and score against each criterion, capturing their evaluation as low, medium or high. This is the scheme used referred to previously. Depending on the complexity of the option, one or two 1½ - 2hr meetings are organised by the project engineer to facilitate this. Scores are captured, along with the justification that the delivery team used to give their final score. The option with the highest rating criterions is regarded as the most favourable option moving forward. A report is written to capture the process, scores and justifications. This allows the decision-making team, which comprises ANSTO senior stakeholders, to consider both the best option proposed by the delivery team as well as the other scored options. The senior stakeholders are then able to have greater insight into the possibilities to pursue with a good understanding of technical risk as well as impacts of each option on strategy and business decisions.

Each application of the BAT on a waste stream generates learnings on the process and methodology of applying the steps. With the application of the process to the HIFAR legacy waste, a couple extra learnings were noted:

1) ***The importance of capturing the delivery team's justifications on each score chosen.*** *An option might partially qualify for two scores. This is a direct consequence of the scheme developed which requires enough scoring options with clear definitions that can be applied, regardless of the concept being assessed. Consequently, the comments attached to the scoring of each criteria become just as valuable to guide the stakeholders in the decision making process as the scores themselves. This also emphasises the importance in developing a universal scoring scheme as mentioned previously, as it would minimise the difficulty of allocating a single score for some otpions.*

2) ***Clear upfront definition on the detail required to characterise the options.*** *An ongoing list of knowledge gaps and uncertainties can be identified in this process. The BAT process should allow for quick decision making based on known assumptions and risks without the need to implement work to gather data for each potential risk identified. The process is a balance of risk appetite and crucial data for characterising the feasibility of the option. A sufficient level of detail that aids the decision making process is only required. Once an option is chosen, detailed work can be commenced on packages of work to address these gaps in order of highest project risk. A performance review is scheduled after an agreed period of time that allows the option to be re-evaluated against the unchosen options to confirm that it remains the best available technique.*

IV. CONCLUSION

The BAT process is internationally recognized and ANSTO favours its use to assess waste management techniques. The

HIFAR legacy waste is one example where ANSTO has been able to implement the process to guide radioactive waste management projects.

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On Selection Of Waste Disposal Packages

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I. Introduction

There is a plethora of waste packages accepted by various international repositories. Each repository provides a Waste Acceptance Criteria (WAC) which dictates the conditions a package is required to meet before it can be approved for disposal in the repository. The stipulations of a WAC will vary based upon the geology, geography, and design of the repository to ensure protection of the environment against accidental release of the contents of waste packages. The WAC forms a critical part of the safety case for a repository. Typically, however, a WAC should provide concession for more than one waste package type to be acceptable. This paper deals only with those packages suitable for disposal of solid waste-forms.

Some international repositories allow for large containers such as customized ISO-standard intermodal containers [1], a commercially available example of which is shown in Fig. 1. Others accept industrial drums of various sizes [2]. Waste producers must consult the WAC of the repository they will be consigning their waste to for identification of their permitted package/s. This paper proceeds on the assumption that a number of waste packages, of different sizes and shapes, are available to be utilized within the WAC being consulted by the waste producer. In general, however, when interpreting the difference between those packages described as 'small' and those as 'large' the simplest comparison is to assume a difference in scale ranging between an industrial drum and an ISO container respectively.

II. Predisposal Management of Packages

The majority of the challenges and costs in waste management are faced at the site of waste production. In addition to having responsibility for the safe disposal of nuclear waste the business operators also hold responsibility to safely manage their waste in the interim. For this reason waste is often pre-packaged and stored in ways that make processing for disposal a lengthy and costly undertaking. This section begins from an assumption that pre-disposal management practices have largely been developed independent of specific knowledge of any WAC.

A. Incumbent Packaging & Infrastructure

Typically waste products will not be initially stored within their final disposal packages at the point of production. This, in part, is due to most industries producing waste well before a repository is available to discern an appropriate final waste package.

If a business tends to produce high volumes of small packages the infrastructure may not be in place to easily begin repackaging into larger, heavier disposal packages. Similarly, if a business already holds large waste packages it may not be feasible to begin to store the contents in smaller packages. Upgrades to infrastructure, or entirely new facilities, may well be required to safely and efficiently process waste into a final disposal package and this cost alone may present the single greatest factor bearing on package choice.

Efficiency of packing the incumbent waste packages, or their contents, into a new package should also be considered. This is where package shape and the limitations of safe handling due to dose become prevalent factors. High efficiency packing may well be plausible, but if the process requires significant infrastructure upgrades or interstitial packages the overall cost to implement high efficiency packing may outweigh the savings this provides. Of course it is every waste producers' responsibility to minimize the volume of their waste as far as reasonably practicable - but what is determined as 'reasonably practicable' is largely dependent on the resources a business has at its disposal.

Further optimization in reprocessing waste into new packages comes in the assessment of the ratios between immobilization media, the waste, and the package. Immobilization media must be sufficient enough to control the leeching of radionuclides whilst not being so voluminous as to dilute the concentration of waste within a package. Lastly, if the incumbent waste packages are not to form part of the final disposal package then the business needs to consider if these may be recovered for re-use, or if they present a new waste stream needing processing.

Figure 1. Croft Design No. 2895 'SAFTAINER' [3].

B. Wasteform Chemistry

The interactions between the waste, any immobilization media, and the package require careful consideration. It may be that a single waste stream requires segregation into multiple types of packages based on the contents to ensure chemical stability of the final product/s.

The size and shape of the package are generally of lesser concern than the material of construction when considering chemical interactions. Whilst a WAC may stipulate a mild steel package as being ample for LLSW it is up to the waste producer to ensure that, if appropriate, a stainless steel package or other is instead chosen. This may require special dispensation from the repository operator to allow acceptance which should be sought prior to waste packaging.

A non-standard package material may not only cost significantly more but will also increase the lead time required to procure the packages. Add to this the time necessary to form a waste stream chemistry management working group to assess the correct package material and the implications of understanding the chemistry of a waste-form have potential to become the critical path for preparing waste for transport and disposal.

C. The Cost of Delay – Escalation & Reputation

Waste management is an expensive and time consuming aspect of nuclear processes. It is also, appropriately, one of the most highly scrutinized aspects of the industry. As such, establishing the correct predisposal management strategy may be fraught with business risk. If a poorly formed strategy is followed through to final packaging of waste in preparation for transport to a repository then the business risks both financial position and safety to personnel in having to retrieve and repackage the waste. To mitigate this risk to its people, and its bottom line, a business may sensibly choose to put off final packaging measures, where safe to do so, until an operational WAC is available. Every revision to a WAC closer to an operational WAC can reduce the risk profile of all nuclear businesses as they become empowered to begin waste management with lower probability of having to repackage. Without a WAC waste producers may choose to delay packaging which poses new risks; one financial and one reputational.

The financial risk is that of escalation; the compounding effect of the devaluation of currency under normal economic situations. Every year of delay in funding critical waste management facilities and programs incurs the cost of a compounding percentage of the total waste management capital.

The reputational risk is an indirect one but represents a significant cost to a business, especially in the Australian context. If a repository operator does not provide a sufficiently detailed WAC far enough in advance of opening a repository then waste producers will not have sufficient time to produce the final waste packages to send to the repository. With a repository being a major and long term piece of infrastructure the public may see waste producers, as well as the repository operator, as having been negligent in preparing for the opening of the repository. Whilst difficult to quantify this risk must also be considered when deciding the time at which a final waste package is decided upon.

D. Commercial Availability of Package

A number of companies provide nuclear package expertise within the international market. Their products have been developed with many of the considerations of this paper as design inputs. These packages, whilst saving significant research and development to the consumer, are costed at a premium. This is especially true for those consumers in Australia who have to bear higher than average costs of freight – and waste packages are not light. If intellectual property is valued by the waste producer it is worth considering if waste packages should be designed in-house, if indeed they have the resources (time, money, and expertise) to do so.

E. Package Dossiers & Waste Stream Management

In repackaging waste for final disposal it is important to consider the traceability requirements on the waste streams within it. The documentation of the history of the waste is a critical factor in the acceptance or rejection of the waste at the repository. Not only will ease of traceability be affected by the volume of waste put into a package but so too will the costs of administering the dossiers for the packages. Sophisticated software packages may be available for tracking waste stream components but will typically require inputs. The greater the number of packages, the greater the volume of data requiring processing and storage. Having more packages, however, allows for easier segregation and it may well be that the comparative volumes of segregated waste will dictate different package sizes for different parts of the waste stream.

III. TRANSPORTATION OF PACKAGES

Choice of disposal package cannot be optimized to solely meet on-site operational parameters if cost effective solutions are to be ensured. A significant cost factor in the disposal of waste at a repository may be the cost to transfer the waste to said repository.

For the reasons following, it is advisable to seek formal quotations from a wide range of freight service providers for the transport of each type of waste package considered when developing cost estimates to support package choice evaluation.

A. Physical Package Parameters

The weight, shape, and size of a package will all directly determine the type of vehicle required to haul the load. Subsequently the type of vehicle required will dictate the routes available to the driver. In this way the optimization of package to the transport vehicle will not only reduce direct cost of vehicle use but also the indirect costs associated with the location of the repository such as asset retrieval as later discussed.

Having a package type that requires only a truck with standard bed dimensions is highly favorable to cost reduction. It is not a requisite, however, as if smaller packages are chosen then it may be possible to load numerous packages onto the same bed shape. This would also allow greater mitigation of package weight variation. With smaller or non-tessellating packages a restraint system needs to be engineered otherwise package certification may be void.

B. Location of Repository Relative to Waste Holdings

The influence of the location of the repository, relative to the location of the waste holdings, will have significant impact upon the freight cost incurred in delivering packages. In the comparative analysis of shipment quotations a broad distribution of costs may be evident - especially if the location of the repository is remote.

When pricing for a delivery the cost to retrieve the company's asset (i.e. the truck and driver) must also be considered. The cost to the freighter to retrieve their asset will depend on variables including the location of the repository relative to their customary supply routes, the size and makeup of their vehicle fleet, transient demand on dangerous goods drivers, and so on. As the truck required to deliver one package type may differ from the truck required to deliver another, so too do the costs of freight based on the choice of package/s.

C. Transport Regulations

Important regulations governing the transport of radioactive materials in Australia include the Australian Dangerous Goods Code [4] and the Code for the Safe Transport of Radioactive Material [5] administered by the Australian Radiation Protection and Nuclear Safety Agency (ARPANSA). Most notably these regulations will dictate the accident scenarios a package must withstand depending on its contents as well as allowable external dose rates.

Packages with a greater radionuclide content may require additional protections. As larger packages may contain greater waste volumes then larger packages may also be required to be built to a higher standard. Not only will the cost of engineering the package be greater, but so too will the cost of certification. Again this must be balanced against the benefit of requiring fewer packages; both in reduced capital outlay and in reduced operational costs to administer the package management.

IV. Consignment of Packages

In Australia it is currently unknown where consignment of waste packages will occur; it is unclear whether the repository operator will provide a transport service where consignment will take place within the waste producer's facility. This presents a risk to all businesses who produce nuclear waste but in particular it remains a significant issue for smaller waste holders who may not be able to facilitate the predisposal management, or maintain the packaging facilities, required to make-ready their waste for transport.

For larger waste producers, such as ANSTO, the waste holdings they are responsible for may make said facilities, and the research and development required to meet a waste acceptance criteria, more economically and technologically viable than for their smaller competitors. For the purpose of this paper the assumption is made that the boundary for the transfer of responsibility for the waste packages occurs at the repository.

A. Repository Operator's Waste Acceptance Criteria

The WAC administered by the operator of a repository dictates which packages will be accepted for disposal and in what condition. The WAC represents the foundational reference piece for the selection of a disposal package and the preparation of a waste-form for disposal within said package.

All of the other parameters requiring optimization discussed in this piece are rendered mute if the package considered does not comply with the WAC for the repository waste is to be consigned to. For this reason an operational WAC allows waste producers to begin to process their waste with engineering surety and financial security. Without knowing that full compliance to the WAC is being pursued in predisposal management, the waste producer accepts the significant safety and financial risk that the packages produced will not be accepted by the repository operator.

Without an operational WAC, waste producers must either risk negligence in engineering; by not conducting full lifecycle predisposal management works, or negligence in financial governance by approving the significant investments necessary to fully prepare waste in the face of uncontrolled risks. In pre-emptively conducting full lifecycle predisposal management of their waste producers may be risking personnel safety and financial burden in potentially having to repackage their waste if it does not align with the WAC once released. This is a critical risk for waste producers to deliberate upon and manage if their repository operator has not released a WAC.

B. Package Inspection, Rejection, & Remediation

Upon arrival at a repository the operator will inspect all waste packages for compliance to their WAC. Furthermore the engineering specifications and quality control documentation will be checked in detail for compliance. If a package is found not to meet the criteria for waste acceptance the package will be rejected by the repository operator. At this time the package, if safe for transport, may be transferred back to the waster producer or undergo remediation work at the repository. In either case this is likely to present a cost to the original waste holder. Where a smaller overpack has been used not only are the costs of remediation likely to be lower, but the cost of transporting the package back to the holding location will also be lower. Whilst a larger package may be more difficult to remediate there is yet another optimization challenge present in that with fewer packages the opportunities for rejection are fewer and more effort can be placed into checking each package upon production for the same operational resource outlay.

C. *Repository Operator's Consignment Fee Schedule*

In Australia there are currently no guidelines to allow the optimization of packages based upon the cost of consignment to a repository operator. For those contexts where consignment fees, if any, are known they should be considered when choosing a package for disposal as the fees, if any, may vary depending on:

- Package size, shape, and/or weight;
- Number of packages within a single consignment; &
- Activity, half-life, and/or radionuclide content of package.

D. *Surplus Packaging*

Nuclear material, waste or otherwise, is typically transported with discretion. For this reason it is unlikely that smaller waste packages, drums for example, will be exposed during transport; they are more likely to be packaged within a larger freight container or covered. This surplus packaging has two cost implications and at least one engineering implication.

The cost of procuring the packaging, and operational cost of placing the waste packages within it, must be considered. Furthermore who will bear the cost of the opposite operation at the repository must be considered. The waste producer should also seek to understand if a repository operator will require the surplus material to be dispatched back to the producer directly.

V. CONCLUSION

The nuclear industry, especially within an Australian context, is highly novel. Novel industry is inherently accompanied by unique challenges. The present example, the choice of a package for nuclear waste, is of particular concern to waste producers within the industry. The safety of staff, the ongoing financial viability of the business, and the social license to operate of the wider industry heavily rely on this choice.

This paper has touched briefly on the broad ranging considerations required to diligently choose a waste package and with numerous waste streams coming to this decision is not a one off process. Some of the most influential factors include the initial process-packaging of the waste, the location of the waste repository, the contents of the operational WAC, and when the investment in the processing infrastructure is made. This shows that this is a decision which is not only not a one off, but one that needs to be regularly reassessed and kept front of mind through the entire lifecycle of nuclear waste materials.

This paper has explored that the safety, environment, and financial satisfaction of business stakeholders relies heavily on the successful choice of waste packages. The choice of package, but also critically when this choice is made, will have significant impacts upon a waste producer as a business. Without properly addressing the fundamental choices of waste packages waste producers will be failing to consider a critical engineering and financial risk to their business.

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Increasing the Technology Readiness of the SyMo Waste Treatment Facility via ANSTO Synroc's Inactive Engineering Demonstration Facility.

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I. Introduction

The ANSTO Nuclear Medicine (ANM) Facility is ANSTO's newly commissioned Molybdenum-99 (Mo-99) production facility. As part of the production process, target plates, irradiated in the OPAL reactor, are dissolved leading to the formation of an intermediate level liquid waste (ILLW). The ILLW is alkaline and contains fission products with the primary source of activity being Cs-137 and Sr-90.

A clearly defined waste management strategy plan for all wastes produced during medical isotope production within the ANM Facility was required as part of the Facility's operational licensing conditions. To treat this waste, ANSTO has designed the SyMo Facility which is currently in construction phase. The SyMo Facility will utilise ANSTO Synroc waste treatment technologies for the termination of the ANM Facility's ILLW stream.

II. SyMo Project

The SyMo Facility waste treatment process involves transforming the ILLW into a highly insoluble product by mixing the ILLW with ANSTO Synroc additives. The ANSTO Synroc additives are tailored to the chemistry and composition of the waste to achieve a final wasteform within acceptable criteria. The ILLW and ANSTO Synroc additives mixture is dried and calcined resulting in a free flowing powder with acceptable properties for consolidation via a hot isostatic press (HIP). The HIPing process utilises temperature and pressure to consolidate the wasteform resulting in a 60% reduction in wasteform volume. The resulting HIPed product is safe for storage and final disposition.

The SyMo process operates continuously in predefined batches allowing suitable intervals for process maintenance to occur.

Due to the activity of the waste being treated within the SyMo Facility, the SyMo process is predominantly constructed within a large hot cell structure. The implication of this is that little to no operator interaction exists resulting in the need for a high level of process automation within the plant.

III. Delivery of the SyMo Facility

The main project risk associated with delivering a first of a kind plant like the SyMo Facility is the translation of research conducted at a modest scale to an industrially relevant scale. Other key risks associated with delivering a first of a kind plant include process integration, process operation, process recovery, maintenance and operator training.

Delivery of the SyMo Facility has relied heavily on a risk management approach ensuring project risks are identified and managed appropriately through the implementation of mitigation strategies to allow for project progression. Mitigation strategies that have been implemented include the development of test-plans, mock-up units and the development of an inactive engineering demonstrator by ANSTO Synroc.

IV. ANSTO Synroc Inactive Engineering Demonstration Facility

The ANSTO Synroc inactive engineering demonstrator is a highly automated 1:1 scale prototype of the process design for the SyMo facility (Figure 1). A need for the inactive engineering demonstrator as a mitigation strategy was identified to address major project risks for the SyMo Project and to increase technology readiness for the final facility.

Figure 1. ANSTO Synroc's Inactive Engineering Demonstrator.

The objectives of the inactive engineering demonstrator are to:

- Test the engineering required for the SyMo Facility.
- Address technology integration risks.
- Test and assess process technology nuclearisation features.
- Develop and test the instrumentation and control philosophy for the SyMo Facility.
- Assess the process operational boundaries and process variability.
- Develop the process quality assurance framework (i.e., to ensure that the product from the SyMo facility met the waste acceptance criteria).
- Provide a facility for the training of plant operators.
- Provide an inactive facility where maintenance procedures could be developed and tested.
- Provide a facility where potential clients could witness the engineering capabilities of ANSTO Synroc.
- Provide a facility for future process development for addressing other problematic and intractable wastes.

A. Process Scaling, Operational Boundaries and Stability

ANSTO Synroc's inactive engineering demonstrator addresses process scale-up risks for the final SyMo Facility. It is a 1:1 scale prototype of compared with the final facility. This has allowed for validation of the process mass and energy balance and to measure accumulation of material within process equipment. This has allowed for validation of maintenance strategies, development of maintenance procedures and verification of the design dose assessment.

Figure 2. ANSTO Synroc's Inactive Engineering Demonstrator.

The inactive engineering demonstrator has also been vital in determining operational boundaries for process units and for the process as a whole. This has allowed key process parameters that influence operation, quality of the product and stability of the process to be identified. Additionally, the parameters at which process instability lies, i.e. the set-points which result in out of specification product, have been identified and are utilised as input to the alarm and response system of the process. Identified operational conditions and long term data on process stability from the inactive engineering demonstrator have been translated to the SyMo Facility.

B. Technology Integration

The inactive engineering demonstrator has provided a platform for the testing of integration of process equipment. Prior to equipment integration, principle process technologies were unitised and tested in isolation. Testing equipment in isolation, with tightly controlled variables, ensured that careful characterisation of the effects of processing parameters (temperature, flow-rate, etc.,) on operation and performance were known and clearly attributable to the parameter under investigation. This ensured that the real effect of process integration (for example the cross-talk between machines) was clearly identifiable and adjustments to process parameters could be made to ensure process stability. The integration work completed in ANSTO Synroc's inactive engineering demonstrator has allowed for the simplification of the SyMo Facility process commissioning as the expected integration interactions have been characterised.

C. Process Technology Nuclearisation

Design, testing and validation of key process equipment with nuclearised features have been allowed for as part of the inactive engineering demonstrator. This includes the incorporation of integrated shielding into process equipment and a focus on the use of highly-modular designs aiding efficient disassembly and reassembly of equipment. This has been driven by the need for remote handling or operator protection during process maintenance within the SyMo Facility. Validation of these nuclearised features using the inactive engineering demonstrator has resulted in lessons learnt which have been translated into the detailed design of the SyMo Facility.

D. Instrumentation and Control (I&C) Philosophy

Testing and implementing a highly sophisticated I&C system allowing for process automation in a robust and reliable manner has been enabled by ANSTO Synroc's inactive engineering demonstrator. Here, process instruments such as flow meters, thermocouples and vibration sensors have been tested for appropriateness, reliability, accuracy, and precision allowing for validation and optimisation of selections.

The current best practice in situational awareness is a key objective under pinning the design of the SCADA system. Here, the operator is presented with only information that is critical to understand the state of the process. This has resulted in the development, testing and validation of performance based graphics that show the status of the process as a whole and at a unit level at a glance reducing operator information overload. The lessons learned during development of the SCADA system for ANSTO Synroc's inactive engineering demonstrator has been implemented to the SCADA system for the SyMo Facilty and will be extended to future ANSTO projects.

E. Process Quality Assurance Framework

Product variability (e.g. inwards goods, composition and properties of the waste and ANSTO Synroc additives steams) has been assessed in the inactive engineering demonstrator as a function of processing conditions. This allows the process boundaries to be characterised that result in the formation of an acceptable product. The results have been translated back to the SyMo Facility and are of high importance as the product produced from the SyMo Facility limits the ability for testing of the quality of the finished product. For this reason, quality assurance must be implemented to ensure the SyMo product adheres to the wasteform performance criteria.

F. Operator Training

The inactive engineering demonstrator has provided a facility in which the facility operators can train in the fundamental chemistry and physics of the process, the general philosophy of operation and process I&C and training for recovery from abnormal situations. This has aided in the development of operational licencing documentation, operational procedures, maintenance plans and procedures, as well as emergency procedures. It has also provided a facility where plant operators and maintainers can be educated in fault finding and rectification on the run.

G. Maintenance

ANSTO Synroc's inactive engineering demonstrator has provided operational data to the SyMo Facility maintainers regarding the frequency of maintenance tasks, the time required for task completion and validation of maintenance dose assessments. The demonstrator has also provided the facility operators with a low-risk environment in which maintenance activities can be performed prior to conducting them in the SyMo Facility. Lessons learnt through maintenance in the inactive engineering demonstrator have been translated into the equipment and process design for the SyMo Facility.

V. Conclusion

As part of ANSTO's strategy to minimise the risks associated with the successful treatment of ILLW arising from the production of medical isotopes within the ANM Facility, ANSTO Synroc has built and operated an inactive engineering demonstrator. This has been highly valuable in increasing technology readiness and design maturation for the SyMo Facility and in ensuring the facility is highly reliable, operable and maintainable. This has been achieved by validating the SyMo Facility's engineering design, developing operational boundaries and addressing key risks including process integration, operation, recovery, maintenance and operator training. The work conducted and the results obtained from ANSTO Synroc's inactive engineering demonstrator have been translated to the design and build of the SyMo Facility ensuring a streamlined build and commissioning process.

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Characterisation and validation of the waste chemistry for the SyMo Facility

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I. INTRODUCTION

The ANSTO Nuclear Medicine (ANM) Facility was recently commissioned to provide potentially life-saving Molybdenum-99 to meet both Australian and international demand. The process begins with the irradiation of target plates in the OPAL reactor which are then transferred to the ANM Facility for dissolution and separation of the Molybdenum-99. Intermediate Level Liquid Waste (ILLW) is produced as a by-product of this process which contains the majority of the fission products.

A clearly defined waste management strategy plan for all wastes produced during medical isotope production within the ANM Facility was required as part of the Facility's operational licensing conditions. To treat this waste, ANSTO is currently designing and constructing the Synroc Molybdenum (SyMo) Facility which will be a first-of-a-kind industrial facility utilising ANSTO Synroc waste treatment technologies to provide final termination of this waste stream.

As part of the development process for the SyMo Facility, an appropriate validated technique to characterise the major components of the waste stream is required. The characterisation technique will determine the chemical concentrations (mol/L) of sodium hydroxide and sodium aluminate. This is a key design requirement for the SyMo Facility as it enables quality assurance of the wasteform product. This will ensure that the wasteform meets the necessary performance specifications and adequately terminates the waste stream with the intention of protecting human health and the environment.

There were a number of key requirements for the characterisation method from the outset of the project. The technique should be selective for the major components of the waste stream, sodium hydroxide and sodium aluminate, along with minimising any potential inferences. The technique should be developed with radiation safety principles in mind, ensuring that the final technique is simple for operators to perform, minimizes sample volumes and manual handling steps. Additionally, the technique selected for the characterisation and validation of the waste should minimise the uncertainty associated with the result. The work targeted an expanded uncertainty of ± 5 % at a 95 % confidence level.

II. LITERATURE REVIEW

As part of the initial work, the literature was reviewed to assess potential techniques that would be suitable for characterisation of the ANM Facility ILLW. The main techniques applied to these types of solutions were volumetric methods, Raman spectroscopy and Inductively Coupled Plasma Optical Emission Spectroscopy (ICP-OES).

The use of ICP-OES techniques for the characterisation of radioactive waste streams is well described in the literature. The general process for this is covered by the standard ASTM C1111 – *Standard Test Method for Determining Elements in Waste Streams by Inductively Coupled Plasma Atomic Emission Spectroscopy*. The use of ICP-OES for this purpose has a number of advantages in that it is readily applicable to a wide range of elements to be characterised and utilises a small sample volume.

However as demonstrated in the ASTM standard, the estimated precision of this technique as measured by replicates of a simulated sample were in the range of ± 6 – 10 % expressed as a relative standard deviation. Additionally, the use of this technique for major elements necessitated large dilution factors and the potential incompatibility of the sample with the dilution matrix must also be considered. Therefore, based on the requirements of the characterisation method set out, it was determined that ICP-OES, without further method development, was not a suitable principle technique for validation of this waste stream.

With respect to the aim of minimising the uncertainty associated with the result, a range of different volumetric methods were selected from the literature as potential candidates for the analysis of the ANM Facility ILLW [1]. These were considered to be direct absolute methods which were directly traceable to SI units by use of an appropriate primary standard. This was not the case for instrumental methods such as ICP-OES.

III. PREPARATION OF STANDARDS

In order to evaluate the suitability of the methods found in the literature, it was necessary to develop a well-defined reference standard which simulated the expected composition of the ILLW from the ANM Facility. This was achieved using high-purity reagents to replicate a similar process to that in the ANM Facility but on a smaller scale.

In the ANM Facility process, the irradiated target plates containing aluminium, low-enriched uranium (LEU) and fission products are dissolved in hot concentrated sodium hydroxide ($NaOH$) solution. The uranium is then filtered out leaving a solution of sodium hydroxide and sodium aluminate containing fission products.

It is worthwhile to note that the aluminate ion ($Al(OH)_4^-$) has an extremely narrow stability range in the $Al_2O_3 - Na_2O - H_2O$ system [2]. Therefore, further concentrated sodium hydroxide is added after processing to ensure that the aluminate present does not precipitate as gibbsite ($Al(OH)_3$). The final inventory of the ILLW is typically 5 – 6 mol/L sodium hydroxide and 1 – 1.5 mol/L sodium aluminate, with an activity in the order of 10^{10} – 10^{11} Bq/L after 3 years decay time has elapsed [3].

The standards which were developed were used to assess the accuracy, repeatability and reproducibility of the assessed analytical methods and are currently used for quality assurance purposes to ensure that the method is in on-going statistical control and no significant degradation of the measurement process has occurred.

IV. EVALUATION OF VOLUMETRIC METHODS

The typical process for measurement of sodium hydroxide and sodium aluminate solutions using titrimetry is some combination of the removal of potential inferences, the titration of the free hydroxide present from the sodium hydroxide and the use of an adequate complexing agent to cause the aluminate ion to release the associated hydroxide ions which are also titrated.

The challenges with many of the methods evaluated were either that they utilised relatively large aliquots or required additional processing steps such as filtration or centrifugation. It was important to develop a method which minimised both the quantities of waste and time that operators would spend handling the material to ensure that radiological exposure was kept to As Low As Reasonably Achievable (ALARA) or Practicable (ALARP) levels.

Therefore, the available literature methods were utilised as a starting point to develop a suitable characterisation method which met the initial requirements. The aliquot volume used for the analysis was reduced as much as feasible. The order of reagent addition was adjusted such that all reagents were added at the beginning of the analysis. All manual handling steps such as filtration were also eliminated from the procedure.

As a result of this experimental work, the final method developed was able to be performed in a single step (see Fig 1.) with the sample aliquot and all reagents dispensed at the beginning of the analysis. Aside from the sampling step, the remainder of the analysis was automated using a commercially available titration system, with all data processing captured in the software.

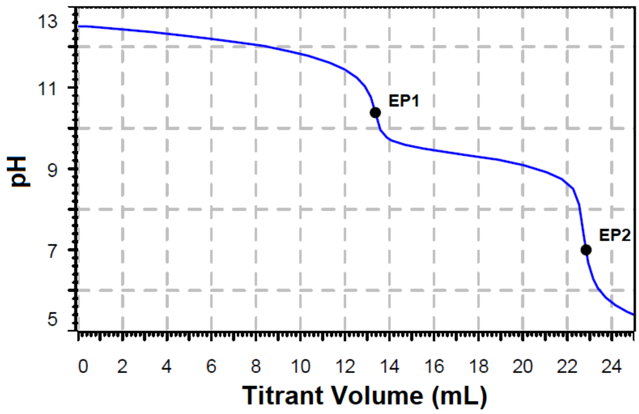

Figure 1. Titration curve for the developed method

V. ESTIMATION OF UNCERTAINTY

The uncertainty of the method developed was assessed by considering a combination of the accuracy, repeatability and reproducibility utilising the developed standard along with considering individual sources of uncertainty throughout the process. These were combined into an overall estimate of uncertainty using the general process described in the *Guide to Uncertainty in Measurement (GUM)*.

$$u^2(y) = c_1^2u^2(x_1) + c_2^2u^2(x_2) + \cdots + c_N^2u^2(x_N) \quad (1)$$

The equation describes the combined standard uncertainty of the measurand as a combination of the individual uncertainties estimated from the analytical method multiplied by appropriate sensitivity coefficients, which describe how the output quantity would be affected by small changes in the relevant input quantities. The combined standard uncertainty is then multiplied by an appropriate coverage factor, k, to obtain a combined expanded uncertainty. For this method, a value of $k = 2$ was chosen to approximate a level of confidence of 95 %.

Based on the experimental data obtained, the estimated expanded uncertainties associated with the sodium hydroxide and sodium aluminate concentrations were found to be in the order of ± 2 – 4 %, which meets the initial requirements set out for the method.

VI. COMPARISON TO DENSITY MEASUREMENTS

In associated work to the development of the waste characterisation method, several simulated waste standards were prepared to cover the full range of sodium hydroxide and sodium aluminate concentrations expected from the ANM Facility. Accurate and precise measurements of solution density were obtained for each of the standards prepared. By mapping the relationships between solution density and the major components of the waste stream, this provided an additional level of confidence in the results obtained from the titration method.

VII. APPLICATION TO ACTIVE WASTE

The developed and validated titration method was subsequently applied to several batches of waste obtained during commissioning of the ANM Facility which contained active material. Based on the analysis, excellent agreement was obtained between the theoretical composition of the waste stream within the variability anticipated due to acceptable variations in the manufacturing process.

The several batches of waste obtained from the ANM Facility were also characterised for solution density. It was observed that the titration results for sodium hydroxide and sodium aluminate corresponded with the predicted solution densities within the estimated experimental uncertainties.

VIII. CONCLUSION

As a result of this work, a volumetric method for the characterisation of the major components of the ANM Facility ILLW stream was developed and validated. The uncertainty associated with the measurements expressed as an expanded uncertainty with a level of confidence of 95 % was in the range of ± 2 – 4 % using an appropriately developed standard.

The technique was applied to active waste from the commissioning of the ANM Facility and the results showed excellent agreement within expected variations in the manufacturing process. The results from the volumetric method were cross compared against the measured solution densities and were found to agree within experimental uncertainty. The technique, which has been developed and validated, will form an integral input to the quality assurance of the wasteform for the SyMo Facility.

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Experience and expertise of the Fuel Company TVEL in decommissioning of nuclear facilities and radioactive waste management

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I. Introduction

Scientists around the world can not come to a consensus on how old our planet is. Humankind has long populated by various nationalities the entire surface of the land. People of various nationalities have long lived all around the globe - from the lands of the far north and to research stations in Antarctica.

However, as the humanity was developing and using the technologies and natural resources necessary for life, the world's population was doing significant harm to the environment, jeopardizing the very possibility of its existence.

Taking this risk into account, in 2000, the UN came up with the initiative to implement in our life and actively promote the Sustainable Development Goals (SDG) and the principles embodied in them [1].

Based on the results of the review, which revealed that global trends in the protection of terrestrial ecosystems and biodiversity, laid down in SDG № 15 (Protect, restore and promote sustainable use of terrestrial ecosystems) the year 2015 (which was initially adopted as the target one), demonstrated a positive development trend. This was characterized by a decrease in the rate of loss of forests, protection and restoration of the key areas of biodiversity and an increase in the amount of financial assistance aimed at protecting biodiversity. However, experts predict that SDG № 15 targets will not be met until 2020, land degradation will continue, illegal poaching and wildlife trafficking continue to hamper efforts to protect and restore vital ecosystems and species [2].

Aware of the looming environmental disaster and the need to provide all kinds of support, not only countries, but also individual companies of different sizes are gradually joining the race to save the planet.

The main threats that jeopardize life and livelihoods are desertification, land degradation and soil erosion. Nevertheless, they can be eliminated with the help of technologies developed in the nuclear industry by the countries of the world.

For example, under the auspices of the International Atomic Energy Agency (the IAEA), nuclear industry companies regularly take measures to assess soil erosion using isotopic methods that ensure high accuracy of the obtained data. This helps to identify and track the centers of soil erosion at an early stage, subsequently reversing the process of land degradation and contributing to soil restoration.

In addition, under the auspices of the IAEA, Member States are provided with the support in fulfilling their obligations to combat desertification. IAEA's support in these areas helps many countries to gather information using innovative methods to create a better agricultural situation for more sustainable land use [3].

Russian Integrator for the decommissioning of nuclear facilities (the subsidiary of the Fuel Company of ROSATOM) is one of the more specific examples how nuclear industry companies contribute to sustainable development.

II. Sustainable Development In Rosatom

Over the past years, ROSATOM has reconsidered its business approaches and begun to focus on the introduction of the principles of sustainable development. The Sustainable Development Goals are becoming an important guideline for the development of the corporate strategy and for the choice of new business areas. In order to adhere to the principles of sustainable development, ROSATOM observes the following rules:

1. As a global market player, ROSATOM recognizes and spreads awareness of the priority of the climate agenda and the importance of the sustainable development principles.

2. Considering the wide range of business lines and products of ROSATOM, the company realizes that it will always influence the achievement of the SDG.

3. ROSATOM focuses on the implementation of the sustainable development agenda including an upgrade of its business and production processes, sustainable development documentation, etc.

4. When new business lines are launched, the company follows sustainable development principles: sustainability of the entire production chain, prevention of environmental damage (focusing on avoiding any damage), ethics, etc.

5. High quality of goods and services that prove competitive on the global market can only be achieved when sustainable development principles are an integral part of the company's business philosophy.

6. ROSATOM realizes the importance of an open dialogue with the stakeholders, as well as the importance of strategic cooperation with its partners in line with the sustainable development agenda.

One of the key elements of the new approach is launching a new separate business line dealing with nuclear back-end activities [4].

III. TVEL – Fuel Company

TVEL JSC holds a leading position on the global nuclear fuel market in terms of fuel production for power and research reactors and icebreakers. Existing data can easily confirm this: TVEL’s share of the Global fabrication market is 17% and its share of the global enrichment market is 36%.

Its key activities include providing nuclear fuel for nuclear power plants of Russian design, expanding this business to nuclear fuel for pressurized-water reactor (PWR) of western design, supplying nuclear fuel for research reactors in Russia and abroad, implementing modified nuclear fuel and technologies, developing non-nuclear businesses in high-tech areas and developing competences in decommissioning and radwaste management.

It is worth noting that TVEL owns about 60% of all nuclear fuel cycle companies in ROSATOM. Therefore, the company can further enhance its expertise in the sphere of decommissioning of nuclear facilities. In addition, TVEL controls the entire production chain for the fabrication of all fuel types [5].

The scopes of activities that are not covered by TVEL are the actual uranium mining (performed by the ore mining division of ROSATOM) and the operation of nuclear power plants (performed by Rosenergoatom).

IV. TVEL – ROSATOM’S DECOMMISSIONING INTEGRATION UNIT

In the beginning of 2019, ROSATOM’s top management made a decision about designating TVEL as the integrating unit for the part of nuclear back-end activities of the corporation.

The Integrator focuses on the consolidation of ROSATOM’s competencies aimed at promoting products and services related to the decommissioning of nuclear facilities (including internal dismantling and associated radioactive waste management) in Russia and abroad.

Currently TVEL can offer the market the following list of services:

A. Safe transfer to decommissioning status;

B. Feasibility study design for decommissioning;

C. Comprehensive engineering and radiation survey, collection of baseline data, surveys;

D. Development of environment impact assessment, Safety Analyses Reports, and related documentation;

E. Development of detailed design;

F. Project management;

G. Decontamination of equipment and buildings on site;

H. Dismantling of equipment and buildings;

I. Radwaste management;

J. Site remediation;

K. Final examination and site release.

In order to organize the decommissioning activities, TVEL will consolidate all competences in this sphere within ROSATOM for the decommissioning activities and engage Russian and international companies with related competences.

The essence of the Integrator’s activity is to carry out the entire process of dismantling of the nuclear facility to the fullest and to eliminate the consequences of its activity. In other words, the Integrator for the decommissioning of nuclear facilities can restore the original ecosystem on a site of an industrial facility of any profile and complexity of activity (even of the nuclear industry), which will be safe for human life and health (SDG № 15). Moreover, such activities help to provide an environmentally friendly attitude towards nature (SDG № 12) by using innovative technologies and accumulated knowledge (SDG № 9). Moreover, conducting its activities on the global market in partnership with other companies, the Integrator provides decent employment and economic growth (SDG № 8) in Russia.

TVEL already has an extensive portfolio in the back-end business and can offer solutions for complex projects. It is worthwhile to note, that under the so-called Federal special purpose program «Nuclear Radiation Safety» for the years 2008-2015, for the years 2016 – 2020 and for the period up to 2030, TVEL has successfully completed a range of decommissioning activities [6].

All activities were completed in full compliance with all Russian legal requirements and regulations. Overall, more than 28 facilities have already been decommissioned.

As set out in the strategy of the company, it is intended to further build up the experience and to expand the range of services in the area. An important driver in becoming successful in the back-end market is building up cooperation and partnerships with other companies in order to gain synergy effects and secure a better world for all of our children and us. Therefore, the company is open to any cooperation suggestions.

V. CONCLUSION

The whole world has embraced the idea of the sustainable development of our planet. The conservation of terrestrial ecosystems is a key principle that underpins the activities of most socially and environmentally responsible international companies with this fact being supported by the above examples from the nuclear industry.

People have already realized that they cannot ignore the urgent issue of conservation of terrestrial ecosystems; however, the approach should be comprehensive, covering a large number of areas of activity and satisfying the sustainable development agenda. ROSATOM and the Integrator are fully committed to this approach. Each of the business areas bears full social responsibility.

The company is acutely aware of the importance and necessity of responsible consumption and production to preserve the ecosystems of the Earth, which is our common home. Therefore, we want to help improve what we have now in order to provide a livable world to our children and pay tribute to our planet.

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The National Radioactive Waste Management Facility project

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I. INTRODUCTION

Australia holds radioactive waste generated from a range of activities over more than 70 years. As of April 2018, the total estimated volume of legacy waste is over 6700 cubic metres (Fig. 1). Given Australia does not use nuclear power, the majority of Australia's radioactive waste inventory relates to the production and use of nuclear medicine and activities supporting nuclear science research and various industrial sectors. The Australian Nuclear Science and Technology Organisation (ANSTO), which operates Australia's only nuclear research reactor, holds over 90% of total waste produced to date in Australia, with similar levels expected into the future. Several other research organisations also hold waste produced from research activities, including the Commonwealth Scientific and Industrial Research Organisation (CSIRO), other government agencies and universities. Currently, these volumes of radioactive waste are held at various locations around Australia.

The National Radioactive Waste Management Facility (NRWMF) will be the first of its kind in Australia, purpose built for disposal of low level waste (LLW) and temporary storage of intermediate level waste (ILW), both current legacy and anticipated future waste. The NRWMF will be one element of Australia's policies for the management of radioactive waste, which are set out in the Australian Radioactive Waste Management Framework. The process for the establishment of the NRWMF is governed by legislation – the *National Radioactive Waste Management Act 2012*.

The facility's design incorporates learnings from overseas facilities and international best practice, as well as local community expectations. It will be an above ground facility with a number of LLW disposal vault complexes and ILW storage buildings. There will also be dedicated buildings or functional areas for administration, waste reception and quality assurance, and laboratories for sampling and testing. The facility site will have several distinct zones including the facility itself, an area containing enabling infrastructure and utilities, an area for agricultural activities and a buffer zone. Technical work undertaken to date has focused on facility concept design, preliminary site characterisation and waste inventory development.

	Low Level Waste		Intermediate Level Waste	
	Legacy	Future[b]	Legacy	Future[b]
Commonwealth				
ANSTO	2,771	4,685	1,211	1,849
ARPANSA	6	36	1	43
CSIRO	1,967	40	419	62
Defence	224	83	60	9
Subtotal C'wth	4,967	4,843	1,691	1,963
States and Territories	8	Not reported	66	Not reported
Industry, hospitals, universities[c]	Not reported	Not reported	13	Not reported
Total	**4,975**	**4,843**	**1,771**	**1,963**

a. Expected waste arising from current or future activities until 1 January 2070.
b. Data reported for these bodies only includes nuclear material held under permit from the Australian Safeguards and Non-proliferation Office (ASNO). There may be other radioactive waste held by these bodies which does not require an ASNO permit, and that is not currently reported in the inventory.

Figure 1. Breakdown of Australia's current and expected future radioactive waste inventory. [1]

II. POLICY SETTING

The Australian Government Department of Industry, Science, Energy and Resources is responsible for administering the *National Radioactive Waste Management Act 2012*, which sets out a legislated process for the establishment of the NRWMF. One important aspect of the process is the requirement for any potential site for the NRWMF to be voluntarily nominated by landowners. The legislation enables the Minister for Resources and Northern Australia to approve and choose a nominated site at his or her sole discretion.

The Australian Radioactive Waste Management Framework sets out the Australian Government's policies for the full lifecycle management of radioactive waste in Australia, now and into the future. By clearly setting out the various elements of Australia's arrangements for the management of radioactive waste, the framework:

- ensures consistency in how waste is managed across Australian Government agencies (as the largest waste holders and generators);
- identifies appropriate accountability for Australia's radioactive waste management practices;
- provides explicit and mutually agreed principles and long-term goals to form the basis of Australia's national approach to radioactive waste policy making;
- provides greater certainty to Commonwealth, state and territory regulators in facility licensing decisions; and

- ensures that Australia's domestic arrangements align with its international obligations.

III. Facility Design

Figure 2. Rendered concept design of NRWMF.

The NRWMF is currently at the concept design stage and will be fit-for-purpose for the disposal of LLW and interim storage of ILW. A rendered design is shown in Fig. 2. The design has been informed by Australian standards, international best practice, expectations of the local communities near the potential sites, preliminary waste inventory development and generic waste acceptance criteria development.

The facility site will be approximately 160 hectares, with about 40 hectares required for the facility itself. Inside the facility, a series of above-ground vault complexes, each approximately 70 metres long, will be used for the disposal of LLW. Each vault complex consists of a temporary roof and a number of concrete vaults, similar to box-like structures, that will be filled with LLW packages. A vault complex will be closed with a concrete lid and covered with a multi-layered engineered earth cap once the vaults inside are full. These structures are shown in the top left quadrant of Fig. 2.

The structures to the right in Fig. 2 depict ILW storage buildings. Each building may contain different types of ILW packages and will have tailored storage configurations according to the waste types. Other buildings in the facility will serve the functions of waste reception, administration, visitor reception and education, research and development, and laboratory testing. The entire facility will be surrounded by a buffer zone and areas for enabling infrastructure and community agricultural research are outside the buffer zone. The LLW vault structures and ILW storage buildings will be built in a staged approach (Fig. 3) because waste will be progressively received at the facility throughout its operational life. The NRWMF is designed to store ILW for several decades while an ILW disposal pathway is developed, and a separate site will be required for ILW disposal in the future. The rest of the facility will operate for 100 years for LLW disposal, followed by decommissioning, closure and several hundred years of institutional monitoring.

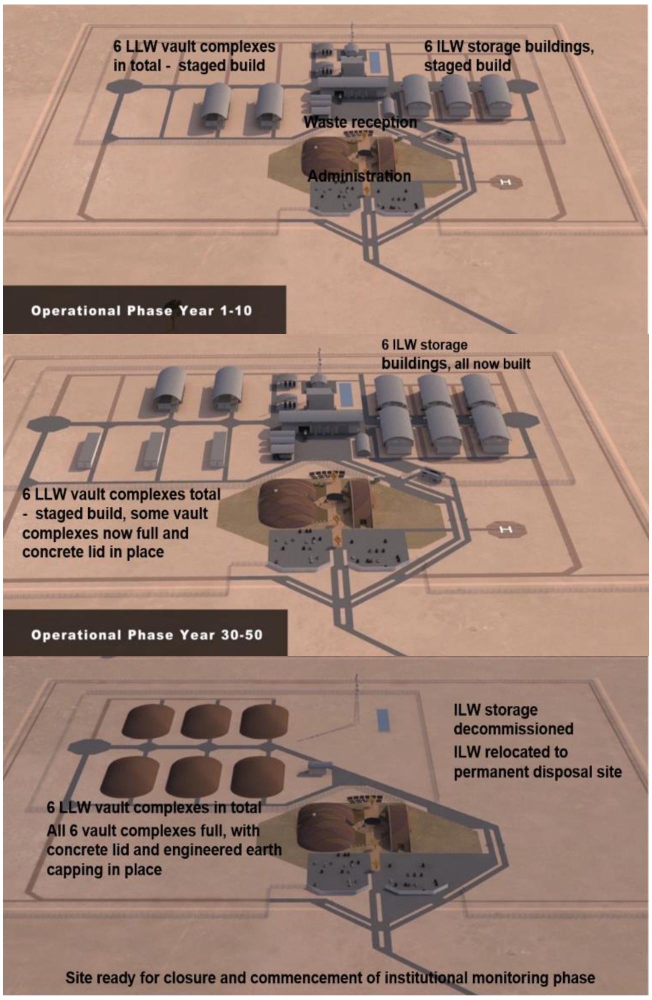

Figure 3. Staged build of LLW and ILW structures.

There has been a significant level of engagement with overseas radioactive waste authorities. This has ensured that international best practice design and methods are considered and incorporated for the NRWMF project. Visits to the El Cabril facility in Spain and the Centre de Stockage de l'Aube facility in France, both of which are managing LLW and ILW in above ground, engineered facilities, have been especially useful for applicable learnings for the NRWMF.

IV. Community Engagement

Engagement with the local communities around the potential facility sites have been especially important because the Australian government is committed to siting the facility where there is broad community support. Ballots run in the communities will help to gauge the level of support. There are frequent meetings of committees composed of community members and a range of educational materials have been published. This helps to ensure that the community members are informed about radioactive waste and the NRWMF proposal.

Extensive engagement has also been conducted with Indigenous groups and Traditional Owners around the sites.

V. Next Steps

Once a site is declared as the NRWMF site by the Minister for Resources and Northern Australia, further works will proceed to determine detailed characteristics of the site. These details will be required to seek approval from Australia's environmental regulator, the Department of Agriculture, Water and the Environment, and nuclear safety regulator, ARPANSA. Design development and waste acceptance criteria for the facility, specific to the chosen site, will also progress. These elements will combine to assist the development of the NRWMF safety case.

VI. Conclusion

The NRWMF will be first of its kind in Australia, purpose built to manage Australia's current and future radioactive waste holdings. Its design incorporates international best practice and continues to progress towards siting licence applications, supported by ongoing site characterisation and community engagement.

Acknowledgment

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TRACK 8: NUCLEAR POLICY, ECONOMICS AND SOCIAL ISSUES

SOCIAL COST COMPARATIVE ANALYSIS TO DERIVE AN OPTIMAL ENERGY MIX PLAN IN INDIA. CURRENT HURDLES AND FUTURE PLANNING

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COST OF CAPITAL FOR NUCLEAR NEW BUILD: THE COMPELLING CASE OF SMRS

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IMPLEMENTATION OF THE FUNCTIONAL-COST APPROACH IN THE DESIGN PROCESS OF NUCLEAR POWER PLANTS

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APPLYING A FUNCTIONAL-COST APPROACH FOR ESTIMATING THE COST OF OWNERSHIP OF NUCLEAR POWER PLANTS IN THE DESIGN PROCESS

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A HYBRID PROJECT MANAGEMENT APPROACH IN THE NUCLEAR INDUSTRY

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NUCLEAR FUEL MARKET MID-TERM FORECAST AND ITS IMPACT TO NUCLEAR FUEL COSTS

A. POLYAKOV
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CONSTRUCTION RISK MANAGEMENT TECHNIQUES FOR DECOMMISSIONING PROJECTS

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SENIOR LEGAL COUNSEL WITH JOINT-STOCK COMPANY ASE ENGINEERING COMPANY, RUSSIA

MANAGING PUBLIC PERCEPTION OF RADIOACTIVE WASTE IN CANADA

SH. WU, R. SIVAKUMARAN, N. PERSAUD
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A LEGAL PERSPECTIVE ON SMR DEPLOYMENT

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WHY IS THERE NO NUCLEAR POWER PLANT IN POLAND? THE PAST AND CURRENT PLANS TO BUILD NPP IN POLAND.

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2 OAKRIDGE SAS, FRANCE

THE ENVIRONMENTAL AND SOCIAL PERFORMANCE OF NUCLEAR REACTOR TECHNOLOGY IN AUSTRALIA IN COMPARISON TO OTHER TECHNOLOGIES

TROY MALATESTA
CURTIN UNIVERSITY, AUSTRALIA

ADDRESING GENDER IMABALANCE: CONTEXT, CASE AND CONSIDERATIONS

BEN STORER
ANSTO, AUSTRALIA

A TOUGH AND BUMPY PATH: THE SUCCESS OF THE FIRST PRO-NUCLEAR REFERENDUM IN TAIWAN

TING-AN LIN
NATIONAL TSING HUA UNIVERSITY, TAIWAN

CASE STUDY ON THE NUCLEAR ENERGY SUPPLY CHAIN EFFORTS AT THE IAEA: HOW THE AGENCY IDENTIFIES PRIORITY MATTERS IN THE NUCLEAR ENERGY INDUSTRY AND DEVELOPS A PROGRAM OF WORK

ANDREW R. CARTAS
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Social cost comparative analysis to derive an optimal energy mix plan in India ; Current hurdles and future planning

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I. INTRODUCTION

INDIA'S orientation in its energy policy has changed dramatically. The target since 2004 was for nuclear power to provide 40 GWe[3] by 2020 with sites like 9900 MW Jaitapur (Maharashtra), 6600 MW Mithivirdi (Gujarat), 9900 MW Haripur (West Bengal), 10000 MW Kudankulam (Tamil Nadu) and PHWRs, FBRs & AHWRs were in the 5-year plan. But every such approved site faced regional protests. The XII 5 Year Plan [2012-17] again envisaged starting of work on eight indigenous 700 MW pressurized heavy water reactors (PHWRs), two 500 MW fast breeder reactors (FBRs), one 300 MW advanced heavy water reactor (AHWR) and eight light water reactors of 1000 MW or higher capacity with foreign technical cooperation. In 2011 NPCIL also aimed for 63 GWe by 2032. This clearly indicates India's ready capability to achieve 63 GWe online from Nuclear source itself. But due to internal introversion, in December 2011 the parliament was told that the more realistic targets were 14.6 GWe by 2020-21 and 27.5 GWe by 2032. In July, 2014, the new prime minister praised India's self-reliance in the nuclear fuel cycle and the commercial success of the indigenous reactors and urged DAE for 17 GWe Nuclear online by 2024 which was earlier envisaged to be 40 GWe by the then prime minister in 2007 with all the revived international cooperations. In May 2017 the cabinet approved ten 700 MWe PHWRs, without locations or timeline, but as a fully homegrown initiative which suggests lower expectations of establishing new nuclear plants with western technology from Areva, GEH, and Westinghouse. Moreover no mention was made of the other elements of the 12th five-year plan for 2012-17, i.e. the Western LWRs which were originally intended to accelerate new capacity additions, and also two FBRs and one AHWR.
"India has nearly five decades of operating experience with nuclear power plants (over 410 reactor operating years) with an enviable safety record. India has indicated its intention to ramp up nuclear power capacity tenfold by 2030 to 63 GW. In the light of India's bold ambition, the National Energy Policy offers the strategy to achieve the target set for the country." (National Energy Policy Draft, NITI Aayog 2017). Moreover for the longer term, the Atomic Energy Commission envisages some 500 GWe nuclear online by 2060, and has since speculated that the amount might be higher still: 600-700 GWe by 2050, providing half of all electricity through its 3-stage Nuclear Power Program.
On the other hand, the government recently doubled down and increased the proposed solar park capacity from 20 GW to 40 GW and is providing financial support to the tune of Rs 81 billion from Central Financial Assistance (CFA) and Rs 6.5 billion from the World Bank[4].
In such a clear scenario of volatile energy planning it is really important to come up with an unbiased optimal energy mix plan. In this paper we would also try to answer this prevalent volatility in energy planning and asynchronous scenario.

II. INDIA'S VAST POTENTIAL & ITS CURRENT SCENARIO

To light homes in the remotest parts of the country and to supply consistent electricity to our Agriculture, Industries and Services sectors, India had set a plan to generate 5000-6000 kWh per capita in which Nuclear, Solar Biomass would play a significant role. "This will sharply reduce our dependence on fossil fuels and will be a major contribution to global efforts to combat climate change", Dr. Manmohan Singh said.
There are vast tracts of land suitable for solar power in all parts of India exceeding 8% of its total area which are unproductive barren and devoid of vegetation. Challenge put to Solar Power (PV type), of not being able to produce electricity during the night time, can be also met by installing pumped-storage hydroelectricity stations. This can be done through the Interlinking of Rivers Plan in India. Canal-top solar power plants when backed up with pumped-storage hydroelectricity points can generate electricity during the night time too.
A 1 MW biogas plant was synchronized to the grid on September 8, 2004 in Punjab[5]. This plant has a capacity to generate 14000 kWh per day (0.58 capacity factor). Additionally, forty-seven tonnes of bio manure, used for agricultural purposes, are produced everyday, which is sold by PEDA at a price of Rs 250 per ton. This 2.42 acre plant supplies electricity @ 3.39 INR / kWh.

Small Modular Nuclear Reactors (less than 300 MWe) and very Small Modular Nuclear Reactors (2-50 MWe) are ready for the fast neutron (FBR) technology. These SMRs & vSMRs may be built independently in off-grid remote sites. Currently, from the total generation of electricity, 30% is lost in transmission, distribution and theft in India every year. Thus these off-grid remote SMRs & vSMRs with very less capital expenditure can also solve this problem. These small reactors without exclusion zones will have minimal land requirements. Four main options are being pursued for these small reactors: light water reactors, fast neutron reactors, graphite-moderated high temperature reactors and various kinds of molten salt reactors (MSRs) as per WNA report. SMR development is proceeding in Western countries with a lot of private investment, including companies with strong entrepreneurial goals, often linked to a social purpose. SMRs, are expected to have greater simplicity of design, economy of series production largely in factories, short construction times, reduced siting costs as per WNA report[1] and passive safety systems. Passive Safety System is free from man-made errors and power-dependency of self-controlled systems in case of any emergency.

India is also soon coming up with its Indian High Temperature Reactor[14]. This 600 MW thermal Nuclear Reactor has multiple advantages:

- Process Heat Generation : 1000 degrees Celsius
- Hydrogen Generation : 80000 normal meter cube per hour.

Hydrogen production using nuclear energy could reduce dependence on oil for fueling motor vehicles and the use of coal for generating electricity. In doing so, hydrogen could have a beneficial impact on global warming, since burning hydrogen releases only water vapor and no carbon dioxide, the main greenhouse gas. There is a dramatic reduction in pollution.

- Electricity : 18 MWe
- Drinking Water through desalination : 3,75,000 liters of drinking water per hour

In India, a cogeneration plant (10200 meter cube drinking water per day MVC) has been installed in Kudankulam supplying drinking water @ Rs.0.5 per litre.

Thus Indian Nuclear Power Program is also ready to solve the transportation fuel demand, drinking water demand and process heat demand for industries.

The per capita annual domestic electricity consumption in India during the year 2009 was 96 kWh in rural areas (136.25 kWh ; 2017) and 288 kWh in urban areas (675 kWh ; 2017) in contrast to the worldwide per capita annual average of 3000 kWh and 6,200 kWh in the European Union.[6]

Currently the overall per-capita electricity generation is 1149 kWh (2018) in India which is 4475 in China (2017), 6,448 in France (2016), 7371 in Japan (2016), 6602 in Germany (2016), 7481 in Russia (2016) and 12071 kWh per capita in USA (2016). Prime sources of electricity generation in India are Coal (196,652.5 MW ; 56.7%), Large Hydro (45,399.22 MW ; 13.1%), Small Hydro (4,506.95 MW ; 1.3%), Wind Power (34,615.1 MW ; 10.0%), Solar Power (24,021.66 MW ; 6.9%), Biomass (8,869.1 MW ; 2.6%), Nuclear (6,780 MW ; 2.0%) and Gas (24,937.22 MW ; 7.2%) as per 30th November, 2018.

Clearly India has a vast potential through its Nuclear, Solar & Biogas Power Programs but in the current scenario the country is far behind its desired target of 5000-6000 kWh per capita electricity generation.

III. METHODOLOGY

To compare different sources for electricity generation Levelized Cost of Electricity (LCOE) is a well known measure. It includes variable fixed - operation and maintenance cost, transmission cost, levelized capital cost. But other factors like Greenhouse Gas emissions (GHGs), Land requirements and Fatalities/Accidents from different sources are just qualitatively mentioned and thus are often missed by the general public and the policymakers in various debates, discussions and energy mix planning. So to internalize negative externalities of GHGs, Fatalities Land Requirements we have derived the marginal social cost of setting up a 1000 MW power plant for all prime sources which will incorporate not only the LCOE but also the other factors as mentioned below.

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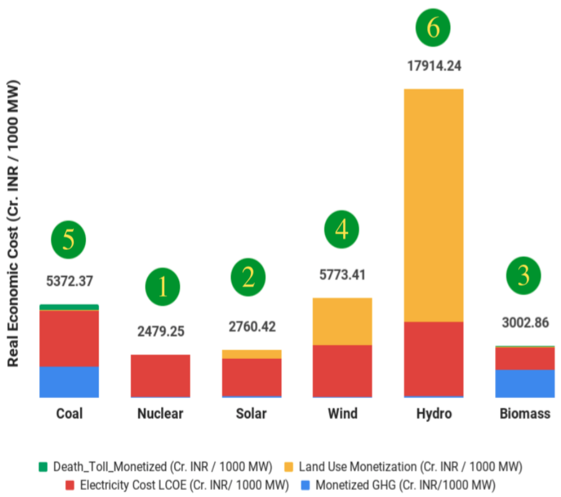
IV. RESULTS......(Not included fully in this template



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Resource	Amount (GW)	Electricity Potential (GW-Years/yr)
Hydro	150	69
Wind (80 m height)	102	26
Biomass Energy	25	20
Small Hydro	20	6
Solar (3% wasteland)	750	150

Fig. 2. shows India's renewable energy sources and their peak-level capacities.

Resource	Reserve (end 2015)	Reserve to Production Ratio 2015
Oil	0.8 million tonnes	18
Natural Gas	1.5 trillion cubic metres	50.9
Coal	60.6 billion tonnes	89

Source: www.bp.com

Fig. 5. shows the available reserves of fossil fuels which will be lasting for 18 Years in case of Oil, 51 years for Natural Gas and 89 years for Coal. This is clearly an alarming scenario.

Energy resource type	Amount (tonnes)	Power potential (TWe-year)
Uranium (in PHWR)	2.22 lakh ton	1.09 - 1.53
Uranium (in FBR)	2.22 lakh ton	58.23 - 196.52
Thorium	8.0 lakh ton	413.33 - 448

Fig. 6. shows that as per 2017, India has a vast reserve of thorium[2] (8 Lakh ton) and Uranium (2.2 Lakh ton) and through its ready 3-Stage Nuclear Power Program can itself generate 3000 kWh for the next 900 years completing the set target of 5000 kWh per capita among 1.5 billion population.

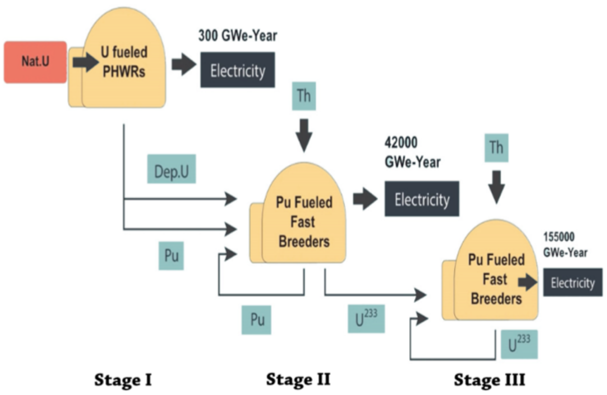
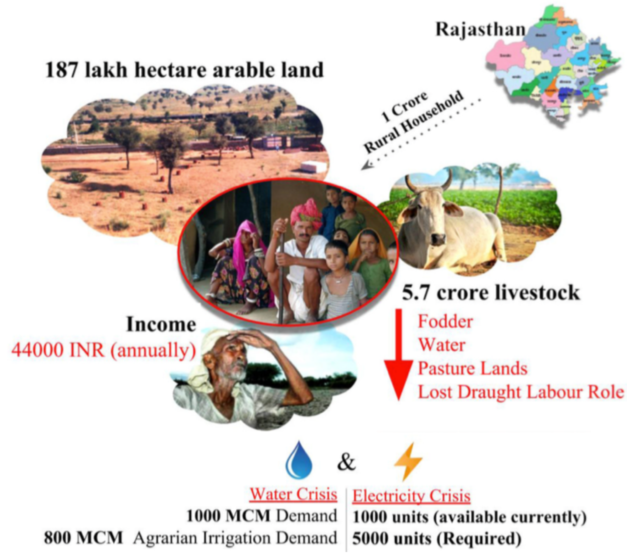


Fig. 7. shows the ready 3-Stage Nuclear Power Program of India. All the 3 reactors (Thermal, Fast Breedor & Th-U233) belonging to the 3 stages respectively can be found in a single city in India i.e. Chennai (Kalpakkam).

Source	Coal	Nuclear	Solar	Wind	Hydro	Biomass
Monetized Indicator (Cr. INR/1000 MW)						
Monetized GHG	1787.8	26.2	95.9	26.2	52.3	1613.3
Electricity Cost LCOE	3241.2	2444.0	2137.4	3031.0	4327.4	1279.0
Land Use Monetization	23.7	8.8	525.6	2715.8	13530.0	33.8
Death_Toll_Monetized	319.7	0.3	1.4	0.5	4.5	76.7



We will also explore the 2 scenarios in Japan and Germany respectively and discover that the Government only with a strong - reliable and sustainable growth target having an unbiased approach to Energy policy and backed up with the required support from the aware citizens tries to achieve the desired targets of the economy.

We hope, our analysis will help India (a multi-party democratic country) derive a socially viable energy plan. We will also show how the currently dead Nuclear Power Plant sites like 6600 MW Mithivirdi (Gujarat), 9900 MW Haripur (West Bengal), 8000 MW Kudankulam (Tamil Nadu), 6600 MW Kovvada (Andhra Pradesh), 6000 MW Markandi (Orissa) and other Pressurized Heavy Water Reactors, Fast Breedor Reactors \& Advacned Heavy Water Reactors can be immediately brought back to commercial operation or construction with pre-project awareness activities (through an unbiased extension network) in the regional public and right evaluation of compensation money in time.

IV. Proposed Planning…...(Not included in this template because Page:limit is 4)*

…Thus the propsoed optimal energy mix plan can be realised in India (a multi-party democratic country) with pre-project awareness activities (through an unbiased extension network) and right evaluation of compensation money in time. This we believe can really solve all the pertaining challenges in front of the nation.…

V. Historical Hurdles Against the major plans of india......(Not included in this template because page:limit is 4)*

VI. Conclusion......(Not included in this template because page:limit is 4)*

...all these hurdles & success stories show that india has been constantly facing these 4 big challenges since decades :

1.) political ambitions
2.) public acceptance (information asymmetry)
3.) lack of pre-implementation grass-root level public awareness
4.) anti-activists/anti-business lobbies

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Cost of Capital for Nuclear New Build: the Compelling Case of SMRs

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I. Introduction

The nuclear industry is struggling to deliver new projects, particularly in western countries [1] where:

a. The industry has lost much of its "know-how" due to decades of inactivity/disinvestment (e.g. Europe, North America…). The consequence being greater execution risks (delays and costs overrun);

b. Electricity is openly traded based on merit order principle (almost) without consideration for (1) system costs (e.g. grid, back-up…) and (2) externalities (e.g. CO2, pollution related disease…);

c. Projects are financed through private investments rather than public one, there is a strong bias against (1) long term and (2) capital intensive projects, typically, nuclear projects.

To expand on the last point, the profitability of capital-intensive projects is highly sensitive to their cost of capital - which reflects the risk of the project and the time value of money [2] and several actors have pointed out the failure of the market to deliver by itself Large Scale Nuclear New Build Projects (e.g. the European Commission [3][4]). While of key importance, the cost of capital is often displayed in the literature as a black-box parameter. The present study details how the cost of capital for nuclear new build project can be reliably estimated together with its practical implications.

In that context, Small Modular Reactors (SMRs) are one of the most promising response brought by the nuclear sector to overcome the above-mentioned hurdles through: production flexibility, reduced size & capital requirement, standardized factory production… The present report highlights how those characteristics materialize in the cost of capital in comparison with conventional large scale reactors.

II. Theorethical Background

By definition, the Levelized Cost of Electricity (LCOE) is the Price of Electricity for which the power project's Net Present Value (NPV) equals zero; and it is namely on this basis that different electricity generation sources can be compared to each other [5].

$$LCOE = Electricity\ Price|_{NPV=0} \quad (1)$$

The NPV of the project is calculated by adding all the Cash-Flows generated throughout the life of the project and discounted at a cost of capital reflecting the risk of the project; namely the Weighted Average Cost of Capital (WACC) [2].

$$NPV = \sum_t \frac{Revenues_t - CAPEX_t - OPEX_t - Tax}{(1+WACC)^t} \quad (2)$$

Where CAPEX stands for capital expenditure and OPEX for operational expenditure, respectively representing investment in physical assets and operational cost such as operation, maintenance and fuel costs.

A project may be financed by a combination of debt (provided by lenders at an agreed-upon return) and equity (provided by the shareholders at an expected minimum return). Because debt and equity are not provided at the same rate, the total cost of capital is obtained by averaging their respective rate weighted by their respective proportion in the project financial structure[1]. The resulting Weighted Average Cost of Capital is defined as:

$$WACC = \frac{D}{D+E} * (1-t) * R_d + \frac{E}{D+E} * R_e \quad (3)$$

Where: D and E are the amount of debt and equity, R_d and R_e denote the costs of debt and equity respectively and t is the corporate tax rate[2].

$$R_d = R_f + (R_d - R_f) \quad (4)$$

Where R_f denotes the risk-free rate (see further) in the corresponding project country and (R_d-R_f) denotes the credit spread of the project, reflecting its credit risk.

In turn, the cost of equity may be determined by the adapted Capital Asset Pricing Model (CAPM). The CAPM methodology builds up the discount rate by summation of several asset-related risk components in order to derive a return at which investors are willing to invest in this asset (e.g. a nuclear power plant) [2][6]:

1 Financing structures range from exclusive equity funding to large debt leverage. Hence, the value of D/D+E varies from case to case.

2 Interests paid on debt are subject to tax deduction. This "Tax Shield" is materialized by a reduction of the cost of capital [2].

$$R_e = R_f + \beta * \left(E(R_m) - R_f\right) + R_{NUC} \qquad (5)$$

Where $(E(R_m)-R_f)$ denotes the equity Market Risk Premium (see further) and β (beta)(see further) is a measure of the non-idiosyncratic, non-diversifiable (or systematic) risk of the project and R_{NUC} is the Nuclear Risk Premium (see further), an asset-specific adjustment factor that accounts for certain non-diversifiable operational risks.

III. Parameters Estimation

A. Estimation of the Risk-Free Rate - R_f

The Risk Free Rate is a parameter that accounts for inflation and the time value of money. Because nothing is really "Risk-Free", long-term sovereign bond rates -in the country considered- are used as a proxy: Damodaran [7] in its database uses Moody's countries rating, Fernandez [8] enquires to thousands financial professionals through annual survey.

B. Estimation of β

Beta (β) is a coefficient measuring volatility, or systematic risk, of an individual stock in comparison to the unsystematic risk of the entire market: beta is a coefficient that quantifies the sensitivity of an industry towards economic cycles. As an illustration, food industry tends to display lower beta values while high-tech industries are more sensitive to economic cycles, which leads to relatively high value of β. The power industry stands in-between. Damodaran calculates beta based on market data of a sample of companies representative of the considered industry [7].

C. Estimation of the Market Risk Premium - $(E(R_m)-R_f)$

The (Equity) Market Risk Premium (MRP) is the premium demanded by investors to be paid over the Risk-Free Rate to compensate for the underlying risk of the project. It changes over time and geography (e.g. tends to diminish for politically stable regions and in time of economic predictability) [7].

Three main methodologies exist to estimate the MRP: (a) through surveys among relevant financial professionals (managers, investors and academics) as performed annually by Fernandez [8], (b) by calculating *ex-post* the effective historical return on the market and (c) by deducing the implied Premium on the market based on current expected companies' dividends. Damodaran [7] exploits the second and third methodology.

D. Nuclear Risk Premium - R_{NUC}

The IAEA "[…] recognizes that a premium does exist, imputable to regulatory concerns, unknown costs, lack of experience, licensing uncertainties, long construction times, concerns about public opinion or public acceptance, and legal conditions in some countries […]" [9].

The Nuclear Risk Premium is a parameter capturing asset-specific non-diversifiable operational risks: risks that another project from the same industry (e.g. the power industry) in the same geography (i.e. same country) would not encounter. Three main categories stand out:

a. **Size** Premium which capture the financial burden of ultra-large project [6][10]. In other words, from a financial standpoint there is a large difference between a single investment in 1GW nuclear project and 1000 investment of 1MW windmill projects.

b. **First Of A Kind** (FOAK) Premium which account for the additional risk that is carried by pioneering projects. It can be either technological or country FOAK.

c. **Political** Premium which translate the risk of asset "hold-up" following a shift in the country political landscape.

IV. Results

A. Reference Case – Large Scale (LS) in Deregulated maket

A reference case must be defined to assign values to the above-discussed parameters. The first case study chosen is an hypothetical large scale nuclear project in fully deregulated market without any state intervention: Europe is used as a proxy. Figure *1* summarizes the situation.

Figure 1. Corporate Financing in deregulated markets

The following table summarizes suggested values for the above-discussed parameters:

TABLE 1. Suggested Value Of Key Parameters

Parameter	Value	Hypothesis/Source
R_f	1,8%	Weighted average of values from [7] and [8]
β	1,5	Weighted average between levered & unlevered total β for Power industry taken in [7]
MRP	5,3%	(2/3)* 20 years average of implied premium [7] + (1/3)* average surveyed MRP [8]
R_{NUC}	5%	4% size premium estimated from model in [11] + 1% due to generally low public opinion in Europe
R_D	8%	Rating (Caa1/CCC+) given the substantial project risk, benchmarked from [3]
D/(D+E)	0,2	Estimated due to lack of publicly available data

Using equation (3) with the value contained in Table 1, comes:

$$WACC = 0{,}2 * 8\% + 0{,}8 * 15\% \approx 13\% \, (\pm 1\%) \qquad (6)$$

At those rates, the corresponding LCOE for nuclear energy would not be competitive with its dispatchable alternatives in most regions (>120$/MWh for nuclear and <100$/MWh for coal)[5]. As a matter of fact, that model has never been implemented. It is worth mentioning that other scenarios exist with various levels of government involvement (Contract for Difference (CfD), loan guarantee…) [12] leading to different levels of WACC reduction [13].

B. Alternative Case – SMRs in a Mature Supply Chain

An alternative is emerging, the Small Modular Reactors (SMRs) [14], sparking hope of a different paradigm.

Because the cost of capital is the arithmetic expression of the underlying project risks, a compelling strategy to reduce the cost of capital would be to suppress or mitigate those at the source.

For the sake a simplicity if we can summarize traditional new build project risks into 4 categories (market, execution, operational and regulatory risks), it appears that the SMR approach would be efficient at inherently mitigating those:

a. **Market** risk relates to uncertainty regarding future operating cash-flow. This specific risk can be tackled by pulling nuclear energy out of the sole production of base load electricity: with (a) flexibility on the output (load following and heat storage) allowed by higher-operating temperature of Gen IV reactors [15]) and (b) versatility on the application (i.e. non-electric such as desalination, fuel synthesis and industrial heat application) also enhanced by reactor operating at higher temperature.

b. **Execution** risk both fed by cost overrun and construction delays but also the financial burden of ultra-large project that may threaten even big companies with bankruptcy. Those risks are softened by: (a) size reduction (of about an order of magnitude [12]) and (b) standardized factory production of critical component such as the Nuclear Steam Supply System (NSSS) and (c) modular construction.

c. **Operational** risk refers to uncertainties regarding the future availability of the plant. This risk may be mitigated by: (1) the reduction of active and non-conventional component failures because of better design simplicity and the implementation of passive safety systems and (2) lower nominal power lessening cold source needs and hence, reducing water access constraints.

d. **Regulatory** risk deals with public concern with potential political “hold-up” and legal & regulatory conditions evolution leading to premature shutdown of the asset. Those risks may be reduced by: (1) fast neutron technology, that have the ability to transmute nuclear wastes; hence reducing the waste burden and (2) passive and inherent safety leading to practical elimination of risk. Both features ease public communication on nuclear energy.

Figure 1. SMRs’ Strategy for Project Risks Mitigation.

The practical implication of those arguments -which is the central claim/hypothesis of the present paper- is that for SMR with co-generation capabilities, once the supply chain is mature, the Nuclear Risk Premium would no longer be relevant anymore. Using equation (3), it comes[3]:

$$WACC = 0{,}4 * 4\% + 0{,}6 * 12\% \approx 8{,}8\% \, (\pm 1\%) \qquad (7)$$

At those rates - and excluding FOAK-related issues- Nuclear Power is competitive with its alternatives [5] meaning that the private sector may take over solely on a market-driven basis.

V. Conclusion

The present study, from a financial standpoint, concludes that, once the supply chain is mature, SMRs would provide a more compelling case than its large scale alternative, everything else remaining equal (e.g. project location). Hence, justifying a lower cost of capital that may, in turn, lead to LCOE values competitive with other energy sources. Private investors would then have appetite to finance SMR new build projects solely on a market-driven basis.

Further work should focus on how to bring SMRs to this level of maturity which is beyond the present scope of this paper. Nevertheless several stakeholders have started investigating the different leads on how to pass through the FOAKs challenges [16].

Acknowledgment

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[3] Considering the European Market

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Implementation of the functional-cost approach in the design process of nuclear power plants

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I. Introduction

The nuclear power plant (NPP) design is a complex process, that requires hundreds of highly educated experts, who develop thousands of systems and make unique technical and organizational decisions on a daily basis. And there are also tens of thousands of different systems impacts, that must be taken into account.

Until nowadays all the functional impacts between different systems were hidden under a large amount of documentation and human expertise. That means, that all the design decisions, made by experts, had a huge amount of human factor. All the initial design data for each designer must be approved first. If a project change happens, all the new data must be reapproved [1]. However, some changes that seem small and insignificant at the first glance, could be left behind, because of the great amount of additional work, connected with rewriting all the technical documentation. As the result, these design changes can have a synergetic effect, that can lead to a number of serious impacts on the entire project configuration. These impacts could cause up to 60% of redesigning works in past, which is an enormous amount of work [2].

These problems are now solved with a new Functional-cost approach (FCA), developed at Rosatom State Corporation company “JSC ASE”. The new approach allows to analyze mutual influence of the NPP elements, as well as to analyze a change process chain, generated by any project cause. As a result, FCA allows seeing all the project decision impacts on the entire configuration and helping to manage them.

II. The Concept of Functional-Cost Approach

FCA is a new design approach [3] that allows controlling NPP lifecycle costs by managing NPP’s individual functions, their design and technical implementation.

FCA is based on functional breakdown structure (FBS) of a project that contains a number of functional links between the configuration elements, so that any design change would be announced to adjacent project elements that might depend on it [4].

III. Use of Functional-Cost Approach to Organizing a Design Process

There are three basic aspects, that must be taken into account when dealing with a NPP construction project costs: Product breakdown structure (“what?”), Functional breakdown structure (“how?”) and Location breakdown structure (“where?”). According to FCA, for integrated project management, the design project should be decomposed to the elements, that are shared by all the three basic aspects [5]. These elements are called functional-cost groups (FCG) and meet the following principles:

- Completeness and integrity of the description of a NPP design.
- Interconnection within the project.
- Tracking opportunity throughout the lifecycle.
- Interconnection between the function and its value.

FCG supply their products (functions) to cover needs of each other and the entire NPP and consist of systems, that ensure their proper work. To support FCG interaction, functional links are created, which together describe the mutual influence of all NPP systems. If one system changes, all the links are analyzed and special impacts are created. These impacts contain all the technical information, that adjacent designers need to take into account.

As the result, there is an impact orientated graph of all the NPP construction project, which is a clear NPP systems impact scheme. It can also be defined as an initial design data handover plan.

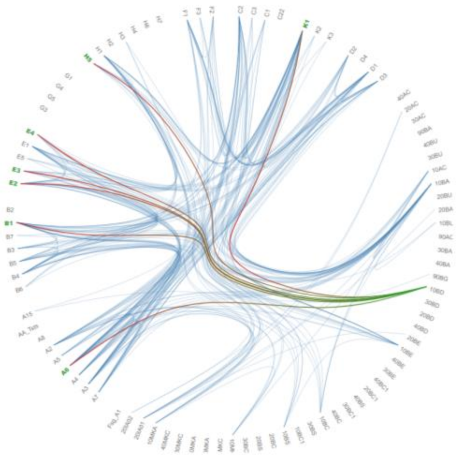

Figure 1. Design project impact graph

Figure 1 shows an impact graph, that defines the project organization structure. Graph node is FCG, which has a designer, who is responsible for its development. A design change of any element may affect adjacent systems, which if they change, can affect further NPP elements. For that reason, a change proposal starts a wave of impacts with technical data, customized for every adjacent designer.

When dealing with a change request from another FCG, a designer must put a list of corresponding systems or decline the change and describe the reason. A design change becomes approved, when all the adjacent designers agreed with it.

FCA approach allows designers to save justification details of each project change in its history, as well as to ensure the traceability of decisions taken throughout the project. In addition, analysis of the sequence of design changes provides information on the impact of the initial design decision on the entire NPP configuration. Examining various design solutions, and passing them through the FCA functional graph, it is possible to distinguish fixed modules of design solutions, which can be further combined into NPP configuration according to customer's requirements.

CONCLUSION

To conclude, FCA is a new design approach, which allows bringing traceability to internal NPP construction project processes. Rosatom State Corporation has already implemented FCA in the form of an IT tool on a number of NPP design projects. Designers have completely changed the approach to their work and now have a clear picture of the functional interconnection of NPP systems; they can spend more time on the substantive part of their work, reducing efforts to write and coordinate internal documentation.

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Applying a functional-cost approach for estimating the cost of ownership of nuclear power plants in the design process

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I. Introduction

Increasing loyalty to nuclear energy in many countries of the world leads to an increase in the share of orders for the construction of nuclear power plants of the main participants, as well as the emergence of new market players. In a highly competitive environment, the customer prefers companies that guarantee the fulfillment of the terms and requirements of the contract, avoid deviations from the contractual cost of the construction, and ensure the low cost of electricity produced by the nuclear power plant.

A balance between the project's capital intensity and the kWh cost can be achieved in the early stages of the project, especially when designing by achieving transparency of processes, monitoring and evaluating design decisions, which are considered not as separate elements, but in conjunction with the subsequent chain of changes.

II. Prerequisites of cost of ownership management in the design process

The issue of ensuring the competitiveness of nuclear energy in the market, along with other types of energy production, is primarily associated with the cost of capital construction of nuclear power plants, which accounts for about 60-65% of the costs throughout the life cycle. Reducing capital and operating costs and, at the same time, increasing electricity production reduces the cost of ownership of nuclear power plants.

The capital expenditure (CAPEX) and operating expenses (OPEX) indicators included in the cost of ownership are calculated and optimized with varying degrees of accuracy throughout the project until the unit is put into operation. The greatest opportunity to influence the values of these indicators exists in the early stages of the project life cycle (pre-contract stage, design), when interdependent design decisions are formed and approved.

The transparency of the mutual influence of design decisions, the relationship of the technical part and economic calculations, control of the cost parameters of individual objects and the project as a whole - all these are necessary conditions for managing the cost of ownership of an object of capital construction in the design process.

These principles formed the basis of the functional cost approach, developed by the engineering company of ROSATOM JSC ASE EC.

III. Applying a functional-cost approach for estimating the cost of owership

Functional cost approach involves the consideration of the nuclear power plant (NPP) project as an integrated system, decomposed into manageable units according to the principle of performing a specific function assigned by the project - functional cost groups (FCG).

The functional-cost approach involves the deep integration of the tasks of assessing the cost of ownership of a nuclear power plant directly into the design process, and its accuracy depends on the quality and completeness of the source data in the form of engineering parameters described digitally in the company's information systems. For the implementation of the functional-cost approach in design, the Cost of Ownership Management Module was developed. In this information platform configuration of the project is formed and constantly updated during the design.

For interconnection of functional groups, functional connections are built between them according to the principle of the supply of resources from related specialties.

In the design process, based on the built-in functional connections, the internal tasks are transferred and agreed upon to supply all the specialties involved in the project in the form of a transfer of technical attributes.

An economic assessment takes place inseparably from the process of exchanging internal tasks and allows approaching the solution of several objectives at once:

1. To evaluate the contribution of the smallest design unit to the cost of the design decision, as well as its impact on the allocated part of the project. The change in the engineering part is estimated by the expenditure (how much will go for its implementation) and revenue (how much will come from the results of its implementation) components. Using pre-arranged connections, you can determine the change in the engineering parts of connected vertices.

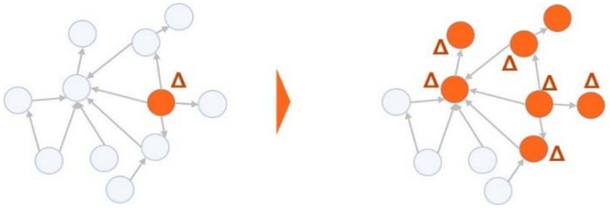

Figure No. 1. Evaluation of the impact of a change in the minimum unit on a project part

2. Evaluation of combinations of solutions and selection of the most cost-effective option. The task is currently under development and will be solved using methods of combinatorics and machine learning.

Figure No. 2. Evaluation of various options and choosing the most optimal

3. Forecasting the cost of the design decision depending on the values of technical parameters.

To evaluate the cost of the design object, it is proposed to use not only direct methods of cost estimation (historical background, monitoring, request to the manufacturer, etc.), but also machine learning methods, which is especially important if there are resource limitations for solving the direct method. Prediction of the cost of design decision, including capital and operational components, depending on the characteristics of the equipment.

Machine learning methods applied to the existing database of equipment catalogs that are in the public domain, a database of technologies and their characteristics used at nuclear power plants, allow us to create dependencies between the cost of equipment and the variability of its parameters. Thus, when changing individual components of the technology or their combination, it becomes possible to predict the cost of the planned design decision.

4. Forecasting the cost of individual parts of the project and the entire project as a whole directly in the design process. Achieving this result will revolutionize the field of value engineering. On the one hand, when designing systems, the designer will be able to select a solution for a given cost, considering a combination of various characteristics of equipment or entire systems. On the other hand, top management will rely on reasonable data on cost when making decisions at the level of the entire project.

IV. Conclusion

It is difficult to overestimate the importance of digital data in the high-tech, business, and manufacturing sectors. For the nuclear industry, researching the accumulated statistical information, obtaining and analyzing new data is a way to exponentially increase the efficiency of nuclear power plant construction projects.

The new approach for the design management, described in the article, allows optimizing the most significant and dynamically changing stage of the project. It also makes the process of making design decisions and changing cost indicators transparent, preserves the integrity of the assessment due to the interconnection of all the functional elements of the project, and defines a new direction for the development of engineering at the junction technical and economic competencies.

Nuclear Innovation: Does Global Policy Keep the Beat with Ambition?

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I. INTRO

Is technological and innovation policy paramount for the success of nuclear power: from long term operation of existing sites to the deployment of new nuclear power builds internationally? Will there be alignment of nuclear energy for sector coupling: industrial process heat to decarbonize energy intensive industries, hydrogen production, desalination and the abundant electricity required for the future transport sector? With greater international attention to small and micro reactors, what role exists for cohesion between policy and driving innovation? What significance should the calls raised by those against the use of nuclear power be factored in the nuclear Research and Innovation (R&I) policy debate including the specific aspect of where to site and construct new nuclear installations - summed up by the terms NIMBY/BANANA1.

In reaction to these enquiries, how do international players and governments design their policies and strategies? Do so-called 'missions' and proposed synergies within the likes of the European Union's upcoming new R&I programme Horizon Europe and the complementary Euratom research programme align to its international endeavours, such as the United Nation´s (UN) Sustainable Development Goals (SDG)? What are the global trends reflecting in R&I policy considering initiatives like the Organisation for Economic Co-operation and Development (OECD)´s Nuclear Innovation 2050, Clean Energy Ministerial Nuclear Innovation: Clean Energy (NICE) Future and the United Nations Framework Convention on Climate Change (UNFCCC) Paris Agreement?

These will be further tackled in the context of areas such as the 'hype curve' of disruptive technologies, geo-political sphere, role of industry and the billionaire philanthropist.

The methodology of this article is to first set the scene in relation to existing nuclear power installations, new build construction and areas of new nuclear beyond the horizon. The links and constraints to adapting to global trends in R&I and the policy perspective is discussed throughout.

II. DRESSED FOR SUCCESS?

A. *Live Forever*

Most of the existing world nuclear fleet was built with a design life of on average 30-40 years. Fast-forward and you will find efforts across multiple reactor technology operators for the delivery of new operating licencing or "Long Term Operation" (LTO) up to 60 and even potentially 80 years [1]. The balance of maintaining systems and equipment for such durations with material aging and radiation embrittlement is one that is intertwined with energy policy. Decisions such as modernising existing systems with *'like-for-like'* replacement parts or new innovative systems or materials is one that is heavily influenced by the economics of such an action and the safety regulatory policy and regime of the location of the plant. The global trends related to adapting new and disruptive technologies, such as those stated in Gartner's hype cycle [2] into LTO programmes come with their own challenges for deployment. For example, the use of some digitalisation technologies, wireless and autonomous control which are very in sync with other major industries are not being integrated into many existing nuclear fleets as part of their LTO, due to a lack in the pace of many regulatory approvals, related system design justifications and qualification costs.

B. *It Takes Time to Build*

After ups and downs of new nuclear build in various worldwide regions over the last couple decades for an array reasons [1]. The current landscape of the nuclear construction projects over the past few years consist of large scale PWR and BWRs. Taking some of these projects through design, licencing, equipment manufacture, construction and commissioning has had some challenges, to put it softly. Regions with a lot of experience such as Europe and the USA have notably encountered cost overruns and delays, while seemingly less experienced regions such as China have not had such issues. Is this luck, chance or a combination of factors the stem from commitment in national and industrial policy, innovation investment and resource capacity development. Its certain there is no one clear factor, but the question remains as a global community have lessons been learnt? Will the First-of-a-Kind (FOAK) banknotes give way to Nth-of-a-kind (NOAK) savings?

[1] *Not In My Back Yard & Build Absolutely Nothing Anywhere Near Anyone*

What is the role global initiatives in nuclear innovation can play in this respect? I would argue the answer to these questions is a positive one and that jointly there are many areas of momentum growing and innovation filtering through due to global R&I policy developments.

III. Electric Feel

A. *Together in "Decarbonised"Electric Dreams*

The world has come together with consensus that the *'future'* - this is seen as *'2050'* in the energy policy debate, should and *will be* decarbonised [3,4]. This means an energy mix that must have a large percentage of nuclear within it must be in place to support the decarbonisation targets – estimates differ per region and analyst [5], but if we take the European Union [6] as a case study it is expected to be close to 15-25% of nuclear in the energy mix in the year 2050. Electricity for normal applications that exist today must be ongoing, even with an energy efficiency policy drive. Moreover, a step change is required to couple sectors that will be heavily reliant on a future low carbon electricity system: transport (electric vehicles and trains), hydrogen production (storage and energy) and to provide direct electricity to energy intensive industries.

There is opportunity now to move with pace in this common endeavour to bring together R&I policy and investment to enable advancement in new advanced nuclear technologies, such as micro and small modular reactors (SMR) [7,8]. However, in the case of regional funding in progressive and collective nuclear R&D, such as in the next Euratom R&D programme the question is raised – is future spending and scope of ambition matched with reality? [9]. Some nuclear industry and environmental stakeholders would argue that not enough is being done to make sure investment and hence R&I actions are driven consistently into all low carbon energy sources on a level playing field manner to make sure the future decarbonisation targets will be met – this can be seen from the ongoing struggles for agreement via the EU *'taxonomy'* developments, [10] which is aiming to set up the framework for a European 'greener' economy and several billions of Euros to transition to it. There is a serious risk to many elements of the future of nuclear R&I is if nuclear power is not considered within the EU taxonomy – such as the message and financial premium it would give to investors who seek to support advanced nuclear technology developments.

B. *The Heat is On*

The international initiatives and cooperation such as OECD´s *Nuclear Innovation 2050*, Clean Energy Ministerial *NICE Future* and the UNFCCC Paris Agreement all elude to the need to look beyond tackling only electricity to fight climate change. Innovation in how the world uses heat from nuclear technologies can be key in this area. District and industrial process heating from large scale nuclear plants has been proven across many sites [11]. But the flexibility and integration of being able to site close to populations due to a reduced emergency planning zone is what many argue will make Small Modular Reactors (SMR)s and Micro Nuclear Reactors (MNR)s more accepted over the coming decades [12]. Therefore, its key to address R&I which will tackle the sector coupling of such reactors and give progress to public concerns and the so called NIMBY/BANANA issue. With investment in R&I that will bring forward nuclear technologies with increased passive safety systems, ability to deliver much higher heat for coupling to industrial processes and even an aesthetic look which is very different from the public perception of what a nuclear site is will enable large strides of progress in public understanding.

C. *Life´s a Gas*

Nuclear coupled to hydrogen production either via conventional electrolysis using electricity or via thermochemical processes that uses direct heat from advanced nuclear reactors are a big benefit being pushed by any SMR developer worth their "molten-salt" [13]. As the reality of what a decarbonised future will entail starts to make traction in the global energy policy and subsequent R&I programmes, it can be expected that the coupling of advanced nuclear to the production of hydrogen will take centre stage in many future debates [14]. However, even though there are various movements within R&I and industry [15] to prove the economic case for coupling of hydrogen production with nuclear technology there is much more required to bring this forward.

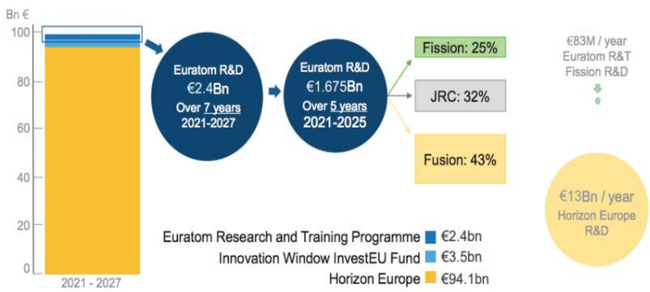

Figure 1. European Union proposed Horizon Europe and Euratom R&I Programme funding allocations and divisions in line with European Parliament recommendation [9].

D. *Smoke on the Water*

Again, on the coupling bandwagon with whatever your SMR flavour of the month would be, is – *'desalination'* [16]. Arguably, the current water resources required for industry and the public will become much more constrained due to climate change in the years to follow. Progress and joint working bridging international policy and objectives such at the UN's SDGs will be key in enabling R&I to bring together sector coupling technology in future SMR deployments in this field. Many SMR concepts are being marketed for their ability to support desalination plants and there have been a lot of stakeholders opening up to the real potential in what this could mean for society – as can be seen from the headline in a 2019

Forbes article "How 1,500 Nuclear-Powered Water Desalination Plants Could Save The World From Desertification" [17]. But the question remains with such opportunity is there adequate global policy support to enable R&I in such areas?

IV. Encore

The role in the 'progress' in nuclear innovation does not lay with any one particular stakeholder – it is for the international community of governments, industry, NGOs and the private investors to work together for a common purpose. Nor does advancement in nuclear innovation have to be labelled '*nuclear'* outright, as many cross-cutting R&I developments will be key tools for use in advanced nuclear technologies. Enablering tools such as advanced/additive manufacturing and digitalisation being considered at the outset of advanced nuclear designs are an example of this. But such enabling technologies will be much greater understood and used only if the nuclear sector and stakeholders are able to participate in pan-national R&I projects – for example within the EU's Horizon Europe and not constrained only to R&I within the Euratom R&D scope (figure 1).

As outlined in this paper the project risks to advance nuclear innovation in the form of SMRs have been reduced over recent years due to strategic R&I investments and initiatives. But its certain that global policy and R&D focussed on enabling nuclear innovation will need to continue to come closer together to make a reality of advanced nuclear technologies to be brought widely to market. The future energy market is going to hybrid, decarbonised and flexible, its clear that nuclear technologies will have a role to play via sector coupling – but there should be no relaxation in global R&I policy development to make this reality sooner rather than later.

In conclusion, the R&I trends in relation to enabling tools and technologies that can be utilised in existing and future nuclear power plants will be a crucial area of progress. The beat of new innovations in digitalisation, manufacturing, hydrogen production, desalination and many other areas need to be continually brought together via international collaborations and programmes in R&I for use with nuclear technologies.

Acknowledgment

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Nuclear fuel market mid-term forecast and its impact to nuclear fuel costs

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I. Introduction

Nuclear power is an important part of the global economy, which generates about 10% of all global electricity. One of the main advantages of generating electricity at nuclear power plants is its low dependence on access to fossil fuels such as oil, coal, gas, etc. The price of electricity for nuclear generation is less prone to volatility in commodity markets. The determining factor for evaluating the economics of a nuclear power plant is LCOE (Levelized Cost of Electricity). In the LCOE structure, the cost for nuclear fuel is only about 14% [1], in contrast to the share of 60% for fossil fuels power plants.

II. Global market overview

World energy companies independently purchase components and services in certain segment of the nuclear fuel market to meet its needs in nuclear fuel:

• the market of natural uranium in the form of U3O8,

• the market for the conversion services of U3O8 to natural UF6,

• UF6 isotope enrichment services market,

• the market for fuel assembly manufacturing services (starting from the deconversion of UF6 to UO2 oxides to assembly of fuel assemblies).

The final price of nuclear fuel which is ready for loading into a nuclear reactor consists of 4 components: the cost of natural uranium U3O8, the cost of conversion services, the cost of enrichment services (separative work units) and the fabrication into finished fuel assemblies.

American market. Total installed nuclear power capacity in the North and South America is about 116,5 GW [2]. The US nuclear fuel market is the largest national market in the world. There are 96 power units in the USA with a total electric capacity of 97.6 GW. Among the main features of the American market, it is powerful to say that the influence of home production is small and a significant proportion of the components of nuclear fuel is imported. The current protectionist measures of the US government, such as the Agreement on the suspension of the anti-dumping investigation of uranium from Russia, etc., also have their role and influence on the market.

Expect of USA also Canada, Mexico, Brazil and Argentina but they have not big influence on the complex nuclear fuel market due to the regional features, such as usage of CANDU reactor which do not demand enrichment services etc.

European market. Total installed nuclear power capacity in the European countries (Western and Eastern Europe) is about 167 GW [2]. After the Fukushima NPP accident, a number of European countries decided to refuse of implementation national programs for the development of nuclear energy although it was previously approved and/or to revise them towards a gradual decrease in the share of nuclear generation in the energy sector breakdown. The main trend in Western Europe in 2012 was the preparation by the number of countries for the future without nuclear energy. Germany, Belgium and Switzerland began preparations for the gradual shutdown process. Notwithstanding to the fact that France and Spain do not plan a complete rejection of nuclear energy, they are faced to new challenges associated with reducing electricity demand. However, there are also countries in the region that continue an active policy of increasing the share of nuclear generation in the energy sector: these are Great Britain, Finland and several Central and East Europe countries.

Nowadays, the European market is characterized by consistently low demand and intensive competition. In the mid-term, the situation is expected when part of the nuclear power plant will be closed, while new ones will not be built, which will temporarily lead to a reduction in demand in the region.

Asia and Africa market. Total installed nuclear power capacity in the Asia and Africa is about 116 GW [2]. The Asian fuel market is the most perspective market is the point of view nuclear fuel components suppliers. The large-scale program for deployment of nuclear power in China, national Japanese strategy of restarting NPP (9 reactors has restarted its operation and 6 reactors have obtained and approval from national regulator as of July, 2019), Korean and Indian market provide the optimistic hopes. After the Fukushima accident Japanese reactors were stopped and the market condition was so that the supply dominates demand. Finally, the chain of governmental decisions, storages and reactors shutdown caused of nuclear fuel market crash.

The natural uranium market is characterized by a relatively high degree of monopolization - about 80% of world production falls on seven uranium mining companies. Among the leading manufacturers of these raw materials are Cameco (Canada), ORANO (France), Rio Tinto (Great Britain), BHP Billiton (Australia), Kazatomprom (Kazakhstan). One of the main trends of recent years on this market is the transformation of Kazakhstan into a leading producer of material, pushing to the second or third places the traditional largest manufacturers - Canada and Australia.

The market for uranium enrichment services is a clearly defined oligopoly, the formation of which is determined by several specific factors, namely:

• historical concentration of technology and production of enriched uranium, which was originally used only for military purposes in several countries;

• sensitivity of enrichment technology to non-proliferation issues and related international restrictions on the transfer of this technology to "nuclear-free" countries;

• a high barrier for the entry of new producers for technological, economic and political reasons.

Currently, there are 4 main companies in the uranium enrichment services market: URENCO (UK-Germany-Netherlands), ORANO (France), CNNC (China) and ROSATOM (Russia).

III. Market Conditions from 2010 till 2019 and Forecast

The scale and pace of development of world nuclear energy sector are the determining factors in the functioning of the global nuclear fuel market. In this context, the consequences of the accident at the Fukushima NPP in March 2011 certainly had a negative impact on the formation of a new image of the global nuclear energy industry and the parameters of new NPP; this process will continue for the next years. National regulatory authorities and NPP operators have implemented the practice of carrying out a set of measures in order to eliminate specific "inconsistencies" identified by the results of stress tests conducted after the accidents in Japan. The speed, completion deadlines and consequences for the nuclear industry are different in each country.

For the immediately preceding period of accident at the Fukushima-1 NPP, the peak of the contracting cycle was achieved. However, in connection with the shutdown of Japanese nuclear power plants and Germany's decision to further abandon the use of nuclear energy, is led to a significant accumulation of uranium products in the storage of energy companies, which had a significant down-pressure effect on market indicators, which is presented in the table 1:

TABLE I. Nuclear fuel cycle components prices (2011-2015)

Price (USD per unit)	**Year**				
	2011	***2012***	***2013***	***2014***	***2015***
U3O8	148.21	127.12	100.31	86.74	96.17
Conversion	10.61	8.56	10.07	7.97	6.91
SWU	149	128.92	108.46	92.33	70.29

TABLE II. Nuclear fuel cycle components prices (2016-2019)

Price (USD per unit)	**Year**			
	2016	***2017***	***2018***	***2019***
U3O8	67.96	57.09	63.82	67.66
Conversion	6.91	5.69	10.32	17.35
SWU	55.29	43.08	36.40	44.8

Calculations provided in the TABLE I and TABLE II above based on Spot Price Indicators published by UxC, LLC in the Ux Weekly and by TradeTech, LLC in the Nuclear Market Review.

The downward trend over several years has had an impact on the "re-valuation" of its economic and production capabilities by major market players. Based on the results of abovementioned "re-evaluation", decisions were made to close Cameco's McArthur River (one of the biggest in the world uranium mining company) and to suspend ConverDyn production (conversion plant), which ultimately led to a significant increase in conversion price indicators from 10.6 USD/kgU to 17.35 USD/kgU. In order to more easily receive information and comply with the paper size requirements, the data is summarized in Appendix 1.

IV. Nuclear Fuel Costs

There are presented assessment of annual nuclear fuel costs per 1 kgU/UO2 contained in fuel assembly. Wide range of using nuclear fuel assemblies and specific design parameters do not allow to use common approach for fuel costs assessment, that is why we have to use cost per 1 kgU in the form of UO2 as the final fission material in reactor. Each operator of NPP has its own strategy of nuclear fuel purchasing and commercial conditions of the contracts with suppliers could differs significantly, due to these circumstances with the aim of forming universal common approach, there some presuppositions which allows to exclude forces of uncertainty. Among such factors may be mentioned the following:

- Usage of mix price indicators of spot and long-term market in different proportions with discount;
- Combination of base price with escalation mechanism and price indicator;
- Usage of fix price mechanism, which allows operators to plan is nuclear fuel costs with great accuracy;

- Partially allocation in binding scope and optional scope of supplies with the aim of hedging the market risk associates with particular contract pricing mechanism

Joint impact of abovementioned forces provides operator additional flexibility in planning of purchases of nuclear fuel components and possibility to catch optimal lowest prices in whole pole of contracts with suppliers.

TABLE III. INPUT DATA FOR CALCULATION

EUP U-235 assay	Natural uranium U-235 assay	Tails U-235 assay
4,40%	0,71%	0,20%

TABLE IV. FUEL COSTS CALCULATION FOR 1 KG U IN FUEL ASSEMBLY (FA)

Costs (USD)	Year			
	2011	*2017*	*2019*	*2023*
1 kg EUP	2 367,6	837,5	1 032,9	1 430,0
1 kg EUP in FA	2 817,6	1287,5	1 482,9	1880,0

Costs for fabrication is equal to 350 USD per 1 kgU as assumption (no MOX-fuel considered) and have not been changed from year to year (WNA provide approximate price for fuel fabrication equal to 300$ per kgU as of 2017).

Calculations performed in accordance with price data provided in TABLE 1 and input data form TABLE 2. With the aim of analysis how annual costs were changed from year to year, we have calculated price for 1 reload (provided to 3 years fuel cycle (1 reload annually).

TABLE V. INPUT DATA FOR COSTS CALCULATION FOR 1 RELOAD

Uranium quantity in FA (kgU)	550
Quantity of FAs in 1 reload	55
Uranium quantity in 1 reload (kgU)	30250

TABLE VI. FUEL COSTS CALCULATIONS FOR 1 RELOAD

Costs (USD)	Year			
	2011	*2017*	*2019*	*2023*
1 reload (mln. USD)	85,23	38,95	44,86	56,80

V. CONCLUSION

Provided information and analysis of these data demonstrated that in 2017 year fuel costs were the lowest in considered period of time. Growth of price indicator from 2018 till this day is an additional confirmation of recovering the nuclear fuel market and provided that growth trend can be save for the next decade with the phase of significant increasing in 3-5 years perspective and stabilizing on plat in 5+ years perspective. If we compare the fuel costs as the changing parameter and other parts of LCOE as the constants, we will get the final comparison table:

TABLE VII. FUEL COSTS AND LCOE

Indicator	Year			
	2011	*2017*	*2019*	*2023*
Fuel costs	85,20	35,92	41,83	53,85
LCOE (%)	14%	6%	7%	9%

The next step of research can be analysis of impact uranium as investment asset to pricing parameters and nuclear fuel market.

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Appendix 1.

NFC / Factor	*ENRICHMENT*	*CONVERSION*	*NATURAL URANIUM*
	General assumptions: • New demand (more than 50 reactors under construction, Japan presumably will accelerate idled reactors re-starts in the face of Olympic games to evade black outs etc) • Pronuclear worldwide movement = some decisions about reactors' outages will be postponed or revised • Supply is shrinking due to unfavorable price situation (McArthur [3], ConverDyn [4] etc.)		
Growth factors	1. Enrichment companies have to make additional investments to replace ageing gas centrifuges for keeping current capacity level beyond 2020 meanwhile: • Current price level well below full production costs • Majority of legacy contracts to expire shortly 2. In 2018 Enrichers declared drawdown of inventories – sign of coming deficit of capacities 3. Market analysis indicate rising quantity of deals with fixation of base price and premium to current price indicators	Shortage of primary supply: • ConverDyn indefinitely suspended production at Metropolis • ORANO faced temporary problems with equipment on Philippe Coste site that caused a 1-2 year delay in ramp up schedule • Chinese producers are expanding capacities in line with domestic demand growth. No export ambitions • Inventory material is being exhausted	1. Deficit of primary supply expected: • Supply cuts by major producers and future closures of elder mines in Australia and Africa • ORANO confirmed the plans of Cominac (Niger) closure [5] at the beginning of 2020th • Cameco suspended mining at McArthur River for indefinite term • KAZATOMPROM may continue production cuts beyond 2020 • Mining costs of perspective projects are considerably higher than current prices (except for Kazakh' and Australian producers) • Investments in new exploration as well as in new projects development are reduced – no new supply expected 2. Low cost U reserves are expiring 3. The role of secondary sources (tails economically viable for re-enrichment, MOX etc.) is diminishing
Decreasing factors	• Phase out policy of Germany (last reactor will be disabled in Y2022) • NPPs are economically challenged by gas and Renewables – risk of demand reduction • Slower pace of Chinese nuclear program as well as Japanese re-starts	• Since 2021-2022 ORANO's plant will operate at design capacity • ConverDyn may resume production at Metropolis any time soon	• Chinese U_3O_8 inventories are high and considering ramp-up of Husab's production China may decrease purchasing of additional U_3O_8 quantities on spot market putting downward pressure on spot prices • US trade issues (232 petition) add uncertainty and slow current market activity resulting in downward pressure on spot prices • Idled projects (Mc Arthur etc) can be put into operation
Forecast	Moderate growth of SWU price in mid-term perspective till **65 USD per 1 SWU.**	Price for conversion continues growing rapidly in near-term with slower growth rate in mid-term perspective till **30 USD per 1 kgU.**	Moderate growth U_3O_8 price in mid-term perspective till **85 USD per 1 kgU.**

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A hybrid project management approach in the nuclear industry

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I. Introduction

The constant development and growth of requirements for the safety and quality of nuclear power plants, as well as the need to comply with the planned dates and cost of building nuclear power plants, require the use of modern approaches to managing complex engineering systems and their interaction.

The costs of building a nuclear power plant make up the bulk of future costs of generating electricity and determine the competitiveness of nuclear power. A significant increase in the timing and cost of building a nuclear power plant may arise as a result of the implementation of various risk events and unplanned changes in the parameters of the course of construction. To achieve the optimal balance between the deadlines and the resources expended, a project approach is used in world practice. However, according to modern research, only 30% of projects are completed successfully.

In this regard, new methods of project management are currently being created and developed, one of which is Agile management [2]. According to statistics, the use of Agile management methods can improve the efficiency of project implementation due to ability to manage changing priorities, increase project visibility, team moral and team productivity [6]. The application of project management in the nuclear industry requires a deep adaptation of existing methods and approaches, taking into account industry requirements, risks and organizational features to ensure efficient and safe operation of complex engineering facilities. Practice shows that Agile management methods are not used for projects implemented as part of the construction of nuclear power plants, due to the variety of ongoing projects and existing tools and practices for Agile management, a large number of risks, and lack of statistics.

An analysis of the scientific and methodological literature showed that currently there are no generally accepted methods for making decisions on the use of Agile management and the selection of the best practices and tools of Agile management for the implementation of projects during the construction of nuclear power plants.

The use of Agile project management methods is not suitable for all types of projects. It is especially difficult to apply Agile methods to manage the construction of complex engineering facilities. In such cases, it is possible to use hybrid project management methods - a combination of Agile and classical management methods [3].

II. APPLICATION OF CLASSICAL PROJECT MANAGEMENT METHODS

Classical methods, project management, by definition, is a mating and unifying activity, and to describe it, you first need to understand the relevant functions, such as engineering support, quality assurance, logistics and accounting. Classical methods relate to all aspects related to the management of the construction of nuclear power plants, and allow commissioning of nuclear power plants as soon as possible, safely and in accordance with high quality standards [4].

Classical methods include leadership functions related mainly to the organization, coordination and control of important tasks related to human resources, equipment and materials, in order to achieve technical excellence, thanks to work in accordance with high quality standards, optimization of the work schedule, supply chain and lower costs. Project management can reduce costs by using a more efficient work sequence, increasing productivity, shortening the duration of work, and reducing the accumulated backlog during construction. The main aspects of project management during the construction of nuclear power plants will be:

- Design Management
- Contract Management
- Stakeholder Management
- Management of risks
- Security Management
- Construction management
- Licensing management

It is obvious that it is advisable to carry out many types of activities sequentially according to well-developed and approved plans, which is the basis of classical management methods

III. APPLICATION OF AGILE PROJECT MANAGEMENT METHODS

Consider the use of Agile methods on a real project for optimizing the technical solutions of Hanhikivi NPP [1]. This work was attended by technical specialists from various design institutes of Rosatom. To optimize the project was given only three months. Given the extremely tight deadlines, based on the accumulated experience, it was decided for the first time in Rosatom to use Agile management approaches. In the optimization work package, the following basic and adapted principles of Agile management were applied:

- main focus is optimization;
- generation and development of all hypotheses, technical ideas and optimization options, analysis and prioritization;
- creating in the Scrum-team the roles of the Administrator (Scrum-master), the representative of the customer, the Technical Leader (Product Owner);
- Formation of autonomous groups of specialists in five areas of optimization;
- planning iteration tasks, daily monitoring, conducting demonstration meetings and retrospectives at each sprint (iteration);
- sprint duration - two weeks;
- rapid introduction and prioritization of necessary changes.

At the beginning of the optimization work, short brainstorming sessions were conducted to identify hypotheses and ideas for optimization. Their results after discussion with a customer representative formed the basis of a basic list of requirements for all optimization work (Product Backlog). At the beginning of each two-week sprint, team leaders drafted a Sprint Backlog, which listed tasks, labor, timelines and priorities for their implementation.

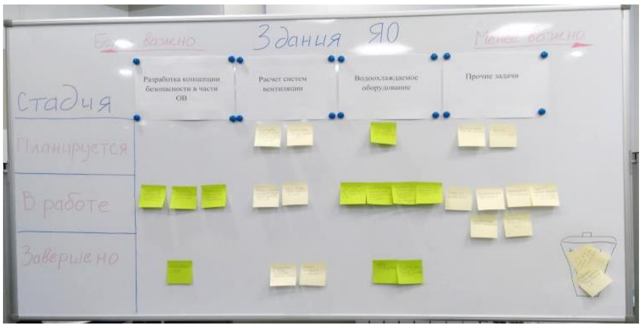

Figure 1. Sprint Backlog.

As a result of a set of optimization work, the main goal of the project was achieved on time — the volume of buildings on the nuclear island of Hanhikivi 1 NPP was reduced by 26%. The number of technological systems and equipment was reduced, the layout of the premises and the general layout structure were optimized, new measures for the physical protection of buildings were proposed. All these results were achieved without compromising the reliability and safety of nuclear power plants.

Among the positive effects of implementing Agile approaches, the following can be noted:

- prompt resolution of any organizational issues;
- joint work of highly qualified experts from various design institutes;
- the possibility of direct interaction with the customer and other stakeholders of the project;
- work of several autonomous teams in one room, high pace of work, team responsibility;
- A disciplining, dynamic workflow structure (sprints, reports, reviews, Scrum tools).

IV. PROSPECTS FOR APPLICATION OF HYBRID PROJECT MANAGEMENT METHODS

Thus, in the framework of the construction of nuclear power plants, it is possible to use both Agile and classical management methods depending on the types of activities. Obviously, a combination of these methods can give the best result [5]. In order to form the best combination, it is necessary to consider the project management tools and choose the most suitable ones. Table 1 presents examples of Agile and classic project management tools.

TABLE I. EXAMPLES OF PROJECT MANAGEMENT TOOLS

№	*Project management tool*	*Agile*	*Classical*
1.	Integrated schedule		+
2.	Task board	+	
3.	Meeting Schedule		+
4.	Retrospective	+	
5.	Reporting form during the construction of nuclear power plant		+
6.	Methodologist role		+
7.	Work in one room	+	
8.	Scrum-meeting	+	
9.	Reporting forms		+

When choosing tools, it is recommended to form the minimum necessary set of tools in order not to complicate the management process. Tools can be selected depending on the characteristics, project risks, labor and financial costs of using the tools themselves. Thanks to this approach, you can independently assemble your own effective management methodology.

Obviously, with the significant advantages of the hybrid project management method, there are also disadvantages. The main disadvantages of the method include the lack of a scientifically-based mechanism for choosing the most optimal set of tools for the hybrid approach. In practice, this can lead to additional project management costs.

V. CONCLUSION

Hybrid project management methods have already shown their effectiveness in practice within the nuclear industry. But for the systematic application of these methods, a scientific justification for the selection of the most optimal project management tools and the formation of appropriate models are necessary. Further actions to develop the applicability of hybrid approaches to managing projects in the nuclear industry should be the formation of statistical indicators and the use of mathematical optimization methods.

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Construction risk management techniques for decommissioning projects

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I. INTRODUCTION

The average age of the worldwide operating nuclear fleet in 2015 was close to 30 years, with nearly 250 reactors more than 30 years old and some 75 beyond 40 years old. While refurbishments for the long-term operation or lifetime extension of nuclear power plants (NPPs) have been widely pursued in recent years, the number of plants to be decommissioned is nonetheless expected to increase in the coming years, particularly in the United States and Europe [1]. The necessity to decommission a large number of nuclear reactors within a short time span raises a number of issues of proper project management.

This article addresses an issue of risk management, which is viewed as a part of project management process. IAEA has recently underlined the importance of this process by issuing a Safety Report "Management of Project Risks in Decommissioning" [2]. The report highlights again differences between the operation of a facility and its decommissioning in terms of risks, safety, training and human resource management, and responds partially to the fact that decommissioning is often undertaken by institutions that lack experience in performing major projects.

The goal of this article is to raise awareness of risk management techniques with the organizations that undertake decommissioning projects and encourage them to turn more often to skills and expertise of the contractors that have experience in nuclear construction. Even in nuclear energy field, which is known to be cooperative and open (to some extent) competitors would rarely share gratuitously their experience in project management. So it is highly desirable for the contractor to have gained experience in fulfilling a number of projects and having learnt some lessons from them.

Atomstroyexport (ASE), subsidiary of Russian State Atomic Energy Corporation ROSATOM, is a contractor in nuclear construction sphere, holding a vast portfolio of construction projects. In December 2018 the portfolio comprised 31 NPP units. It is the author's belief that some information from this article might be insightful for the institutions active in nuclear decommissioning.

II. DECOMMISSIONING AS A RISK MANAGEMENT FIELD

To define the challenges a contractor faces in a certain activity, it is necessary to define the activity itself. Nuclear decommissioning is often understood as "the administrative and technical process whereby a nuclear facility such as a nuclear power plant (NPP), a research reactor, an isotope production plant, a particle accelerator, or uranium mine is dismantled to the point that it no longer requires measures for radiation protection" [3]. Another common definition is "all clean-up of radioactivity and progressive dismantling of the plant [4]".

A dedicated IAEA Safety Guide outlines that "the term decommissioning refers to administrative and technical actions taken to allow removal of some or all of the regulatory controls from a facility ... These actions involve decontamination, dismantling and removal of radioactive materials, waste, components and structures" [5].

The author has also found that a number of other definitions are in use. In the essence however they are all consistent with the IAEA definition, and the latter will further be used. Moreover the article will be restricted to the ground-based facilities of civil use (NPPs and research reactors in the first line).

By definition nuclear facilities decommissioning involves dismantling and deconstruction (or demolition) of buildings and structures, which makes it similar to construction in nature. Therefore the author stand on the ground that the competencies necessary to perform these activities are mainly those that are usually expected from construction companies.

Operators, who are most often facilities or research institutions, gradually learn that, although they employ excellent and appropriately skilled operating personnel, it becomes essential either to bring in companies that have decontamination and dismantling as their core service and expertise or build up such competencies themselves which can turn costly and time-consuming.

A decommissioning project is an endeavor which bears much risk both in the terms of technical performance and in the terms of cost and time constraints. Therefore organizational aspects, including project and risk management, and the organization of work, personnel, knowledge and competences, are of the essence in the execution of decommissioning projects.

Organizational requirements emerging in decommissioning projects are similar to those of a construction project, and differ substantially from those relevant to the operational phase of a nuclear facility. Project risk management, including time and cost management is much more significant than during routine operations. At the same time, the risk profile changes from nuclear to industrial safety, because spent fuel is removed from the site. However the residual radioactivity still requires high protection of the workers and the public from the hazards of ionizing radiation, which in turn requires better quality management.

The obvious difference of a nuclear decommissioning project from its "conventional" counterpart is the involvement of radioactive material (and in some cases also nuclear material – when the spent fuel is not removed by the operator for some reasons). Although a slight difference, it entails huge consequences in the manner, in which the works have to be performed, and the safety demands.

Both decommissioning and environmental remediation are major industrial projects in which the safety of the workforce, the local public and the environment must be ensured from both radiological and conventional hazards. The underlying key requisite of the decommissioning activities is to ensure the long-term safety of the public and the environment, and the continued health and safety protection of decommissioning workers.

This makes the necessity of disciplined risk management obvious for such projects. As set in [2], many of the risks specific to decommissioning projects arise from the circumstances which are not usually found in ordinary operation of the facility, such as non-routine and first of a kind activities; ongoing requirements to deal with unknown conditions; an uncertain working environment etc.

III. WHAT IT TAKES TO INTRODUCE RISK MANAGEMENT

Risk is regarded as an uncertain outcome that can negatively or positively affect the achievement of organizational objectives. Considered in this manner risks either impose threats or offer opportunities. Primary intention of risk management is therefore regarded as maximization of opportunities and minimization of threats. It has to be noted however that in a conservative framework of construction projects risks are more often considered to be threats so that the actions associated with risk management tend to be more protective than exploratory as the case would be with opportunities.

Risk management has long found its way into management of complex projects. The basic document giving an outline of risk management procedures and providing principles of such is International standard ISO 31000 [6] (introduced into Russian regulation as GOST R ISO 31000-2010), which endorses formal risk management process, embedded into project management routines.

This formal approach seems to be the only way of disciplined risk management process, but it also may be a hindrance to other processes. Risk managers (like some other professions) are often considered as a nuisance by management, who make things seem overly complicated and intimidating.

Introduction of risk management can (and more often than not will) face opposition, as it suggests considering threats and opportunities differently, not in the manner usually exercised and therefore unnecessary. Therefore the greatest threat to the risk management process itself is reducing practice-based and benefit-bringing job to a mere form-filling exercise. This effect depends greatly from the corporate culture but the author dares observe that bureaucratic companies are most prone to falling to this inadequate way of managing project risks.

Such practice can reduce to zero any positive impact that risk management brings. The author regrets to mention that there is no simple recipe to challenge that state of things.

However adopting risk management framework can bring a number of benefits as described in [2]. To put the benefits short, proper risk management ensures that the decisions are taken wisely and resources are allocated where they need to be, with greatest attention given to areas where a threat realized can harm human lives or health, make disastrous negative environment impact or render the project impossible in terms of initially stipulated costs and timeframes.

Although additional workload may feel redundant by management, as stated above, some paperwork is inevitable in the risk management procedures. In ASE risk management procedures are set by the Organizational Standard, which is a regulation document to be used in all company's projects. The standard is compliant with ISO 31000:2009. Alterations made to ISO in 2018 have yet to be implemented in the organizational documents. Risk management therefore forms an integral part of the broader framework of the corporate procedures, such as decision-making, budgeting, communications, contract management etc. and project management documents.

The author considers this to be a good practice as it allows risk management not to be regarded as a separate process standing away from the main course of project fulfillment. With regard to this paper however, it becomes highly impractical to consider the document in question in full detail, as it will mean boring the reader's attention with the whole organizational and project management system of ASE.

IV. DOCUMENTS ARE IMPORTANT …

Bearing that in mind, the author would like to describe the great role one document, risk register, plays. It proved to be the most consistently used instrument and forms both an instrument of risk management at operational level and a basis for a yearly report which helps gaining expertise and keeping experience for use in future projects.

By definition, the primary objective of risk management at the operational level is to control risks during the implementation and execution of a project. In ASE all risk management activities are divided into "planning procedures" and "operative procedures" (compare to the difference between terms 'risk management' and 'managing risk' in [6]), the latter being active remedial actions will not be described in any detail.

The core in planning is filling a risk register with identified risks, evaluation of the risks possible impact and responsible persons (more on them below). As proposed in [2] the project risk register serves as the record keeping tool for capturing all of the relevant details for each of the identified project risks. The risk register allows for day to day tracking of the risks and helps in prioritizing the risks and in developing the action plans for which the project team has responsibility.

Risk register is also a base for (and technically may form a single electronic file with) the general risk management action plan, which in turn describes activities and measures for managing the respective risks.

ASE is a project-oriented company and its risk management procedures are project-based. The project risk register is by rule initially populated at the phase when the EPC contract with the ultimate customer is being negotiated. The negotiating team then transfers the register to the project implementation team.

Planning procedures of risk management are performed in iterative fashion and are conducted quarterly, semiannually and annually based on the level and severity of the risks. The results of monitoring also find their way into the register. Such process makes the register (combined with the general risk management action plan) a "living" instrument, which is turned to in decision-taking on a regular basis.

Risks management report, which describes realized risks, preventive and mitigation actions, etc. is drafted annually to be reviewed by the management of the respective project and becomes a valuable source of information on good practices and improvement areas.

V. … BUT MORE IMPORTANT ARE THE PEOPLE

A special person – risks coordinator – is assigned from the project implementation team. This person bears the primary workload of risk management together with the risks owners. The author finds it a good practice to allocate a special person to keep track of risks and coordinate all related persons. It should be avoided however that this person is considered detached from the bulk of the project team. Parallel can be made here to quality management, which too should be a common effort, not a standalone activity.

At the stage of risk identification the risks are assigned to one of the subject categories (such as technical, regulatory, political, legal, HR etc.)

After the risks are identified, "risk owners" are assigned to each subject category. These persons may or may be not actual experts, who took part in risk identification. Normally they are members of the project implementation team, or part of the relevant in-house supporting teams (such as a legal counsel for a group of legal risks). This person together with the risk coordinator becomes responsible for risk management activities in his or her respective field. In particular risk owners do the following:

- estimate risks manageability and define strategies for managing them;

- plan preventive and response measures;

- define triggers for enacting response measures;

- ensure that the introduced measures do not cause undesirable effects and secondary risks;

- appoint responsible performers, who are supposed to immediately implement the planned measures;

- parameter residual risks.

Risks are being monitored by risks owners too, and where it regards critical risks (in terms of combined possible impact and probability) they are monitored not rarer than on a monthly basis.

Risks coordinator consolidates individual risk management measures into the register / action plan as described above and tracks the actually implemented activities and their results for the annual report.

VI. TAKEN TO PRACTICE

The author would like to use this article to share a story of one risk in ASE practice being mitigated through the use of the risk management practices mentioned above.

Being a lawyer of an NPP construction project, the author is in the position to describe a risk, which may seem to be coming from the contract management realm. Since contracting and contract management procedures are similar in the construction and decommissioning projects, the author believes it to be appropriate for this paper.

The risk in question is the risk of losses due to inconsistencies between terms and conditions of the ultimate customer contract and those of the contracts down the supply chain. This risk is an obvious one, so it was identified rather early and made its way to the risk register and assigned to the project lawyer as the risk owner.

Transferring the risk to the subcontractors by "mirroring" the terms of the main contract was the decision of choice and it was mentioned in the action plan part of the risk register.

However it quickly became clear that such decision would raise the cost of the subcontract.

Therefore other options needed to be found to mitigate the risk in question. The actual activities belong to contract management techniques and are not covered by the topic of this article but the main idea is that all proposed and taken actions were noted in the risk register. Moreover, transfer of risk ownership took place in cases of some subcontractors.

The project is not over yet and not all lessons to be learnt are taken from this particular example, but the author's personal belief is that the described techniques not only bear benefits to project management, as initially envisaged, but also help to discipline the approaches of the project team members and can help to organize the work within such teams.

VII. Conclusion

The author would like to emphasize again that reaching project management endeavors (such as nuclear object construction or decommissioning) craves for project management techniques. One of such techniques is risk management.

The author's personal experience gained from a number of construction projects is that risk management is a useful technique, which can bring large benefits on both operational and strategical levels, unless it becomes bureaucratized and has no attention from the decision-makers.

The author finds that integration of risk management procedures into general procedures is one of the numerous keys to success and designating a coordinator to consolidate contributions of risks experts and risks owners is another one. And it comes without saying that maintaining a decent risk register combined with an action plan and some form of "extracting" experience from these efforts (e.g. an annual report) is the primary condition for any management procedures.

It was not the author's aim to give a comprehensive description of risk management as such and to describe all procedures which are relevant to the case. Large zones have not even been touched here, such as definition of risks basing on their severity or ways of managing specific risks, or implementation of the findings into decision-taking process.

It is the author's hope however that this article will attract some attention from the facilities involving decommissioning companies and that risk management instruments and techniques described in this paper will help avoiding threats and reaping opportunities in any decommissioning project.

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Managing Public Perception of Radioactive Waste in Canada

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I. Introduction

Nuclear waste has emerged as a very controversial issue in the Nuclear Power debate. The public perception of the final disposal of radioactive wastes is a significant concern for long term nuclear waste management. The concept of Deep Geological Repositories (DGR) was introduced in Canada as a potential solution and various organizations are engaging the local communities to inform them and garner support.

Canadian nuclear waste is sorted into one of three general categories, low, intermediate, and high-level waste. The Canadian Nuclear Safety Commission (CNSC) is the federal regulator for all nuclear activities in Canada. It enforces the Nuclear Safety and Control Act which is the basis for low and intermediate level waste disposal strategies at all licensed facilities. The Nuclear Waste Management Organization (NWMO) was established to manage the long term used fuel (high level waste) storage plans as per the framework outlined in the Nuclear Fuel Waste Act [1].

As of June 2018, Canada had an inventory of about 2.9 million used nuclear fuel bundles, with about 90,000 additional used fuel bundles being generated each year [5]. As more Canadian nuclear reactors undergo life extensions through refurbishment, the question of what to do with nuclear waste becomes all the more pressing. Nuclear waste disposal has major repercussion for the nuclear industry because without a long-term waste storage solution, the viability of the industry is uncertain [1].

Members of the public can play a prominent role in nuclear energy policy, and it's noted that the support is generally correlated with their level of experience and knowledge of nuclear energy [6]. Considerable efforts are required by proponents of DGR to sway public perception. Canadian organizations such as the CNSC and NWMO are employing different strategies for public consultations and risk communications to win the perception war.

II. Current Landscape

Low level waste consists mainly of industrial items that have become contaminated with low levels of radioactivity during routine maintenance at nuclear generating stations. Intermediate level waste is more radioactive and consists primarily of used reactor core components. Currently, low and intermediate radioactive wastes are stored above ground on site by the owners. It may also be transferred to an authorized waste management operator such as the waste management facility operated by the Canadian Nuclear Laboratories (CNL) [1]. Sometimes the low-level waste is sent off to companies like Energy Solutions or PermaFix for volume reduction.

Canada's used nuclear fuel (high level waste) is now safely stored on an interim basis at licensed facilities located where it is produced, such as Bruce Power and Ontario Power Generation (OPG). The used nuclear fuel in Canada is also held onsite in interim storage facilities and is the responsibility of the nuclear operator. The NWMO is responsible for implementing the approach to the long-term management of Canada's used nuclear fuel [1].

When used nuclear fuel bundles are removed from a reactor, they are placed in wet storage where their heat and radioactivity decreases. After seven to ten years, the bundles are placed in dry storage containers. Figure 1 shows the Canadian facilities that manage used nuclear fuel on site [1].

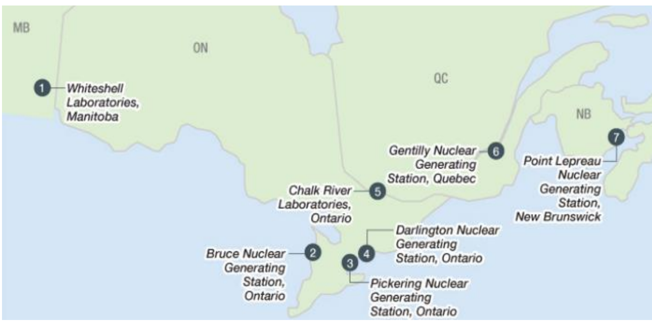

Figure 1. Canadian Nuclear Facilities

Like many other countries with nuclear programs, Canada is planning for the future. The NWMO uses a host application process to find communities that are willing to host a repository for spent fuel with support from the local government. These potential hosts are then evaluated to determine the best location. Creating a plan for the long-term, safe, and secure management of used nuclear fuel is an important responsibility for the protection of people and the environment. The NWMO has met with thousands of people from many parts of Canadian society to hear their concerns and feedback on the plan [2]. The NWMO has engaged the local communities, elected representatives, Aboriginal peoples, technical and social subject matter experts, environmental and faith groups, and businesses about many social, technical, economic, environmental, and ethical issues involved.

III. Deep Geological Repository (DGR)

While we can continue to safely store nuclear waste above ground, we also have an obligation to future generations to dispose of waste permanently and responsibly. The DGR would safely isolate and contain waste underground, ensuring the protection of water and the environment. The current proposed location by Ontario Power Generation (OPG) for low and intermediate level waste is at the Bruce Power site [7]. As shown in Figure 2, it will be deeper than the CN Tower is tall and constructed within low permeability limestone capped by 200 metres of low permeability shale. The site has rock formations over 450 million years old that do not have any major faults or factures, are stable and predictable, and provide excellent isolating capabilities [3].

The storage dept is more than 680m. The proposed DGR will be located more than a kilometer away from any part of Lake Huron and will have no impact on the surrounding environment. The solid rock formations around the DGR will limit the movement of radioactivity to extremely slow rates. In addition, there will be multiple natural barriers of solid, stable rock protecting Lake Huron from the DGR.

The NWMO is leading a site selection process for Adaptive Phased Management (APM) for spent fuel. It is drawing on the studies completed for the OPG DGR. The process for selecting a site for a used nuclear fuel repository is actively continuing in study areas around several communities in Ontario.

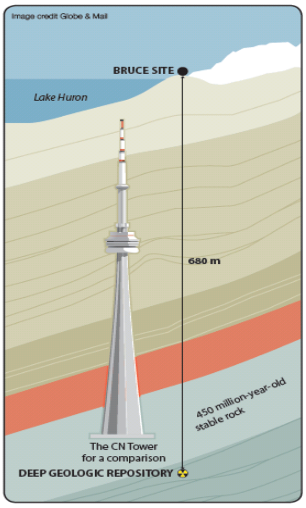

Figure 2. Illustration of the approximate depth of a DGR

IV. Challenges with Public Perception

Over the years, concerns have been raised in the media, by the public and other intersted groups in relation to the nuclear industry and, in particular, its waste. Some commonly expressed views are that the radiation hazard from nuclear waste is permanent and unmitigable, and it is a danger to people and the environment. The opposition believes that there's no acceptable permanent solution for nuclear waste; even if waste is contained within a DGR, it can infiltrate the rock and cause harm over time.

These concerns are impeding proposed nuclear waste management projects. During public consultation of the OPG DGR, the most common concern was that radioactive material might seep through the rock and contaminate the lake [10]. A member of the public pointed out that there is no existing design that has been proven to be safe in the few attempted DGRs. In contrast, the Environmental Assessment Report [8] produced by the Canadian Environment Assessment Agency stated that it would take water about 10 million years to move just one metre, and the rock in the area has been undisturbed for a million years. This makes a leak from a DGR extremely unlikely. First Nations around the proposed site have also raised questions on having nuclear waste buried nearby [9]. They worry that the DGR might reduce the spiritual value of the land or infringe on their treaty rights. Although these concerns do not have a scientific basis, they must be addressed because public acceptance is essential for project success [10].

For example, in Belgium, projects for the implementation of a repository failed due to the absence of the public support. The Belgian authorities developed and implemented a methodology for waste processing. All the international technical recommendations were incorporated into the project scope and as such, they believed that the strong technical basis of the project would be enough for the implementation of the strategy. However, in all the locations chosen during the preliminary review, the local communities refused the government proposal. The short list of locations was evaluated against technical criteria and deemed ideal; hence, the authorities did not anticipate any opposition nor conduct any public hearings. This failure led the agency to realize that establishing waste disposal infrastructure would inevitably have perceived economic, social and ecological consequences. Based on these lessons learned, stakeholder engagement with the candidate communities was incorporated into the implementation strategy and their concerns were addressed. Starting from 1998, projects began to use this approach, considering the interests of the community rather than relying on the technical merit of the project alone [3].

In the United Kingdom, members of the public criticized the nuclear industry for not being open and refusing frequently to supply relevant data. The British nuclear industry created Nirex, a company whose purpose is to examine the financial aspects of safety, environment, and DGRs for storing low and intermediate level radioactive waste. It recognizes that frankness and transparency are necessary to obtain trust. To overcome the possibility of local opposition, the public engagement programme was organized by the University of Bristol's South West Nuclear Hub to address key issues such as energy, waste disposal, and health effects. They provided the public with an insider's perspective on the operation of a nuclear power plant. The team consists of experts in health physics, safety cases, fuel transport, operations, and materials science. The emphasis for these experts was on engagement and discussion rather than technical communications. Several approaches have been taken to extend the impact of the public engagement programme including school visits to educate young people, large festival exhibitions, media coverage, and discussions with special interest groups. Together with industrial partners, this programme utilized educational outreach and public engagement to improve the public's understanding, trust and acceptance of the nuclear sector [3].

Finland currently has the world's first DGR for spent fuel under construction at the Olkiluoto Nuclear Power Plant on the west coast of Finland. The site selection process started in 1983 and a construction license was finally issued in 2015. The plan is to bury used nuclear fuel around 400 meters deep. The waste will be packed in sealed copper canisters before being transferred into tunnels and further into deposition holes lined with a bentonite buffer. The Finnish Nuclear Energy Act facilitated high participation from local stakeholders in the decision-making process. During their Environmental Impact Assessment, nuclear operators must obtain a "Decision in Principle" from the government, ratified by Parliament), that affirms the project benefits society as a whole. In 2001, the Finnish Parliament ratified the Decision in Principle on the Olkiluoto facility [12].

V. INPLEMENTATION STRATEGY

Under the leadership of the NWMO, the current plan for Canada is to have the used nuclear fuel be contained within a DGR in a suitable rock formation. Used fuel will be isolated from people and the environment using a multi-barrier system.

The NWMO has set out a set of principles that reflect the value, concerns, and priorities of Canadians.

- *"Focus on safety – Safety, security and protection of people and the environment are central to the siting process. Any site selected must address scientific and technical site evaluation factors that will acknowledge precaution and ensure protection of present and future generations and the environment for a very long period.*
- *Meet or exceed regulatory requirements – The outcome must meet, and if possible, exceed all applicable regulatory standards and requirements for protecting the health, safety and security of humans and the environment.*
- *Informed and willing host community – The host community, the local geographic community in which the facility is to be located, must be informed and willing to accept the project. The local community must understand the project and how it is likely to be impacted by the project.*
- *Right to withdraw – Communities that decide to engage in the process for selecting a site as potential hosts must have the right to end their involvement in the siting process at any point."*

NWMO's plan is called the Adaptive Phased Management (APM). It is the result of a three-year dialogue with both specialists and the general public [2]. It also takes into consideration the best practices adopted by other countries with nuclear power programs. The NWMO is committed to collaborating with all interested and potentially affected individuals and communities. This includes communities that expressed interest in participating in the siting process, First Nation and Metis communities, and surrounding municipalities. The site selection process started in 2010 with 22 municipalities and Indigenous communities that expressed interest in learning more and exploring their potential to host the DGR. The NWMO has gradually narrowed the focus to a few locations through technical site evaluations, social engagement to assess safety, and the potential to build supportive and resilient partnerships. The Township of Ignace in northwestern Ontario, Township of Huron-Kinloss, and the Municipality of South Bruce in southern Ontario are considered potential host communities.

VI. CONCLUSION

The destination of the waste generated in a nuclear power plant remains a big challenge. It is incredibly important to engage with the public. The nuclear industry must continue to communicate its benefits and promote itself in a more transparent and straightforward manner. There is a strong correlation between the concerns of the public and the lack of information about radioactive waste [11]. The concerns are greater in the places where the information is inadequate or insufficient. The examples studied here show that the public perception can be a vital factor in the successful implementation of radioactive waste storage solutions. When not considered, the experience demonstrates that this can cause serious setbacks for the project. Some will always oppose the nuclear industry, but the goal is to find a community that is engaged and welcoming.

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A Legal Perspective on SMR Deployment

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I. Introduction

This paper will provide a legal and regulatory perspective on the deployment of small modular nuclear reactors (SMRs). It will overview the lessons learned from prior nuclear sector construction experience and assess the needs, and potential impact, of regulatory infrastructure on future SMR projects. In conclusion, it is suggested that all lessons from historical and more recent nuclear power plant construction should be reviewed and actively incorporated into new SMR projects and that the implementation of optimal regulatory regimes and licensing processes will be essential to facilitate successful SMR deployment.

II. What are SMRs?

Small modular reactors or "SMRs" are newer reactor designs commonly understood as having electric power up to 300MW. Common characteristics of SMRs include modularization and advanced safety features. However, SMR technologies differ widely in terms of power range, coolant, applications and intended deployment scenarios. The heterogeneous nature of SMRs needs to be kept in mind when considering regulatory frameworks and licensing approaches.

III. Considerations for SMR Deployment

Setting aside issues of policy, public acceptance and energy planning, one of the primary impediments to deployment of SMRs is the investment of financial and human resources necessary to take a design from concept to commercialisation. This task is particularly challenging with so many SMR designs being conceptualised and the need to overcome significant regulatory barriers to market entry.

Prior to SMR deployment, customers must seek the approval of the host country's nuclear regulatory authority. Of course, this is the case for the conventional large reactor technologies, as well as SMRs. However, with many SMRs presenting regulators with novel approaches in design, safety systems and/or deployment scenarios, the licensing process, at least for first-of-a-kind projects, could be lengthy and costly, particularly where extensive modelling, testing and validation are necessary.

Historical and recent large reactor construction projects have continued to demonstrate that regulatory frameworks and licensing processes impact commercialisation costs, project schedules and project budgets. For SMRs, they have the potential to be particularly impactful where primary drivers of project economics are linked to standardised designs and multi-unit deployment.

Customers and vendors must also develop the commercial structures and negotiate the contracts through which to finance and develop their projects. Of course, their joint aims should be the successful deployment of one or more SMRs on time and on budget. However, the challenges that have arisen in the more recent wave of large nuclear power plant construction around the world demonstrate that that nuclear sector projects are far from straightforward. As the industry formulates and implements future SMR projects, all impediments experienced in the sector to date should be carefully reviewed.

IV. Lessons Learned From Prior Nuclear Construction

Upon studying historical and more recent conventional nuclear power plant projects, in particular the Generation III+ units, and identifying the factors contributing to project delays and cost overruns, the patterns and similarities in these projects are striking.

Some of these factors include:

- *Government:* Political risk and changes in government policy can significantly impact the outcome of a project. High levels of government commitment and projects in which government-owned and/or government-supported companies are the main sponsors and participants are likely to be more successful and better able to withstand unanticipated project delays and cost-overruns.
- *Regulatory*: Regulatory frameworks and licensing processes can impact commercialization costs, project schedules and project budgets. Certainty and transparency are vital, as is robust communication between regulators and licensees.
- *Change in law:* A "change in law" (which includes a change in regulation, as well as change in interpretation and application of law and regulation), can significantly affect the implementation of a project.
- *Human resources:* Experienced human resources in key roles throughout design, engineering, construction and commissioning, are necessary. Their absence can lead to project delay and cost overrun.

- *Project management:* Experienced project management – for both contractors and owners – is essential for project delivery. Owners need to be ever-mindful of their obligations as licensees, including regulatory compliance and ensuring that they are equipped to act as an "intelligent customer".

- *Contractual basis:* Fixed-price "EPC turnkey" contracts can place tremendous pressure on contractors, as well as the overall project, and ultimately be disadvantageous for owners. In particular, privately-owned contractors and owners may struggle to withstand liabilities and escalating disputes.

- *Project schedules:* In hindsight, there are many recent examples of unreasonable and overly-ambitious project schedules, particularly in the context of first-of-a-kind (FOAK) construction projects. Care should be taken to develop realistic and achievable schedules, including taking potential regulatory uncertainties into account.

While SMRs undoubtedly raise new issues for project delivery, it will be prudent for all stakeholders to take into account the lessons learned from historical and more recent nuclear power plant construction and seek to ensure they are not repeated. This is, in particular, the case for FOAK SMR projects.

V. Legal and Regulatory Framework for SMRs

Changes to and the development of new laws and regulations require thorough review and analysis, with significant investment of time, expense and human resources. The challenges are increased where cross-border issues arise and multiple regulatory regimes are potentially relevant.

In established nuclear countries, regulators must consider whether and, if so, how existing legal and regulatory frameworks applicable to large nuclear reactors need to be modified for SMR licensing. Newcomer nuclear countries contemplating SMRs face the multiple hurdles of developing an appropriate SMR legal and regulatory framework and undertaking the first in-country SMR licensing process through a nascent regulatory institution with limited human resource experience in these activities.

The structure of an SMR licensing process and some of the regulatory requirements that will underpin it warrant consideration, in particular, to realise projected SMR benefits and appropriately address some of the innovations presented by SMR designs.

For example, the following issues need to be considered:

- *Multi-module facilities:* Multi-module facilities raise numerous regulatory issues requiring resolution.

- *Off-site/foreign country manufacturing:* There are regulatory implications for significant offsite manufacturing, fabrication and fueling/refueling.

- *Export controls:* Export control regimes should be reviewed for export of fully fabricated and fueled units from the supplier country/countries to the host country.

- *International transportation:* International transportation standards and requirements may need to be re-considered for transportation of fully fabricated units, fully-fueled units and units containing spent fuel.

- *Deployment scenarios:* Safety, security and non-proliferation considerations arise for below grade reactor designs, floating SMRs, SMRs anchored to the seabed and SMRs sited in other remote locations.

- *Nuclear liability:* Principles of international and domestic regimes governing third party liability for nuclear damage should be reviewed to ensure they are suited to SMRs.

- *Passive safety systems:* Challenges of licensing new passive safety systems need to be managed and overcome.

- *EPZs:* Determinations need to be made concerning sizing emergency planning zones around SMRs and siting closer to human population centres.

- *Control rooms:* The operation of multiple SMR units from one central control room by an individual operator needs to be studied and results demonstrated.

- *Decommissioning:* Decommissioning requirements, including funding schemes, for projects where modules are physically removed from the site need to be determined.

Some of the above issues may be applicable to all SMR designs, while others may be design and/or deployment scenario-specific and for which bespoke approaches may need to be developed. It is possible that regulators around the world may treat and answer the above questions in different ways, potentially being influenced by historical approaches to nuclear regulation and licensing. Clearly this would generate additional complexities to be navigated by SMR vendors and customers contemplating international deployment.

VI. Conclusions

In deploying SMRs, we need to ensure that we learn from historical and recent nuclear sector construction experience. There are lessons to be learned at every stage of the process, from designing a regulatory regime and establishing a licensing basis, to procurement and contracting, and to implementation of the construction phase. Many of the lessons demonstrate the significance of the regulatory regime and licensing process on commercial project outcomes.

We currently have an opportunity to embrace the challenge of applying lessons learned to date and developing new or modified regulatory regimes to simultaneously maintain responsible nuclear and radiation safety while facilitating advantageous technological innovation and successful commercial SMR deployment.

ACKNOWLEDGMENT

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Why is there no nuclear power plant in Poland? The past and current plans to build NPP in Poland.

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I. INTRODUCTION

Poland has tried to build a nuclear power plant for decades. The first attempt to construct a NPP was undertaken in 1982, but the project was finally abandoned in 1990. The reasons were changes in political and economic regime, resulting in problems with financing as well as antinuclear protests of misinformed public. Many years later, Poland faces a challenge of reducing its dependence on hard coal, which is a great undergoing for the country being second largest coal producer in Europe, with 79% of its electricity coming from this fossil fuel. In 2009, nuclear program is back on the table, bringing the hope for GHG emissions reduction, clean air and diversification of energy sources. Since then, the plans and schedules have changed several times. Now, the first Polish NPP is expected to go on-line in 2033.

The purpose of this paper is to provide a historical overview of the NPP plans in Poland. The struggles of a country being a black spot on the map of CO2 emissions and pollution, in its path towards clean energy, energy diversity and security will be presented. The questions: why there is no NPP in Poland and why Poland needs it, will be discussed.

II. POLISH ENERGY SYSTEM

The capacity installed in the national power system was close to 46 GW at the end of 2018 (over 36.6 GW are professional power plants based mostly on hard coal and lignite; over 6.6 GW is professional capacity installed in renewable energy sources, the rest are industrial power plants (various fuels) - approx. 2.7 GW). Electricity demand is approximately 171 TWh per year [1]. In the next several years (especially after 2029), a significant part of the currently operating generating units will be withdrawn from the system. Poland is second largest coal producer in Europe and a black spot on the map of CO2 emissions (figure 1).

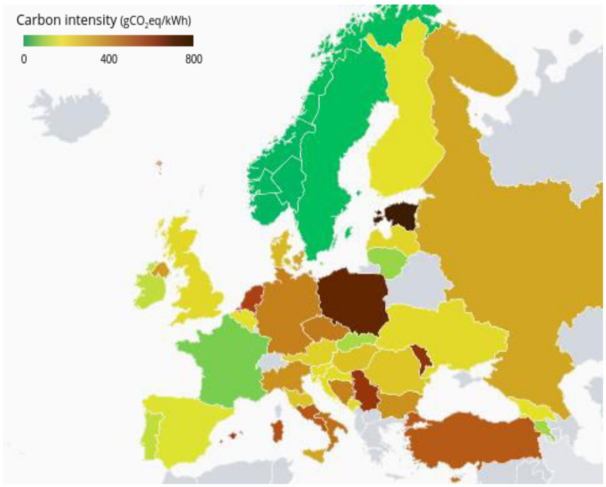

Figure 1. CO2 emission in Europe [2]

III. EUROPEAN UNION CLIMATE POLICY

The European Union (EU) has long been committed to international efforts to tackle climate change and felt the duty to set a significant pressure on changes in energy sector policies in European countries.

In 2009, the 2020 package (the so-called 3 x 20% package) was enacted. It sets three key targets [3]:

- 20% cut in greenhouse gas emissions (from 1990 levels);
- 20% of EU energy from renewables;
- 20% improvement in energy efficiency.

In 2014, the EU maintained the direction of combating climate change. The European Council approved four goals in perspective 2030, which after the 2018 revision they have the following shape [3]:

- 40% reduction in greenhouse gas emissions compared to 1990 emissions;

- at least 32% share of renewable sources in gross final energy consumption;
- increase in energy efficiency by 32.5%;
- complete the internal EU energy market.

The so-called Paris Agreement is a key to current policy and activities. During the 21st Conference of the Parties to the United Nations Framework Convention on climate change (COP21) in 2015 it was said that people need to stop the growth of the global average temperatures below 2°C compared to per-industrial levels and pursue efforts to keep it to max. 1.5°C. The aim is lead in global climate action and to present a vision that can lead to achieving net-zero greenhouse gas emissions by 2050. On 28 November 2018, the Commission presented its strategic long-term vision for a prosperous, modern, competitive and climate-neutral economy by 2050. It is important to make a socially-fair transition in a cost-efficient manner.

IV. FIRST POLISH NUCLEAR POWER PLANS AND ITS DOWNFALL

First nuclear power plant plans came to light in 1970s. In 1982 Council of Ministry decided to build a nuclear power plant in Żarnowiec, with the first reactor was to be finished in 1989 and second reactor - in 1990. Technology chosen was VVER 400, but the total capacity of the plant was supposed to reach 1600MW. Still in 1987, the Polish government was planning second NPP in Klempicz. It was envisaged to construct one unit every 2 years up until 2000. The first nuclear program planned 6MW from nuclear, however the changes in political systems after 1989, economical aspects as well as activist protests, were some of the reasons why Żarnowiec NPP construction was shut down [5][6]. The construction of any other NPP then Zarnowiec NPP has never started.

V. NUCLEAR RENAISSANCE

In twenty first century, first resolution of the governent calling for nuclear power was in 2009. Polish Energy Group Nuclear Power Plant 1 (PGE EJ1) was created in 2010 as a company managing the investment. First version of the Program envisaged building 2 NPPs with joint installed capacity 6000 MWt was proposed in 2011.

There project faced several problems on the way. Our goverment spent two years for international consultations. Last country to agree with the undergoing was Austria in 2013. The problem was also with siting analyses – there was contract breaking with WorleyParson company in 2014 due to little progress from their side. Schedule was updated in 2017 and is followed by now, with the works currently focused on finalization of siting process and building public acceptance.

VI. CURRENT PLANS

A. *Energy Policy of Poland until 2040*

The near-term future of Polish fuel and energy sector is based on several strategic documents like Poland's energy policy until 2040 (PEP2040). It provides response to the most important challenges facing the Polish energy sector in the coming decades and sets development directions of the energy sector, including tasks necessary for implementation in the short term. The goal of the state energy policy is energetic security, while ensuring competitiveness of economy, increasing energy efficiency and reducing the sector's impact on the environment, simultaneously providing optimal use of domestic energy resources [1].

In 2018, PEP2040 announces new plans - to build first nuclear reactor around 2033 and later to add one unit every 2 years. The goal is 6 reactors in 2043 with the installed power 6-9 GW [1].

These dates result from the energy mix in the national power system. Without additional investments in new energy sources, it is at this time that further losses will occur in covering the increase in demand for new capacities resulting from phasing-out existing generating units, especially based on coal. NPPs will reduce domestic greenhouse gas emissions and air pollution (both CO2 and others, e.g. NOX, SOX, fine particulates) from the energy sector.

Installed net capacity by technology unit and electricity generation by technology unit, which are estimated until 2040 are shown in figure 2 and figure 3.

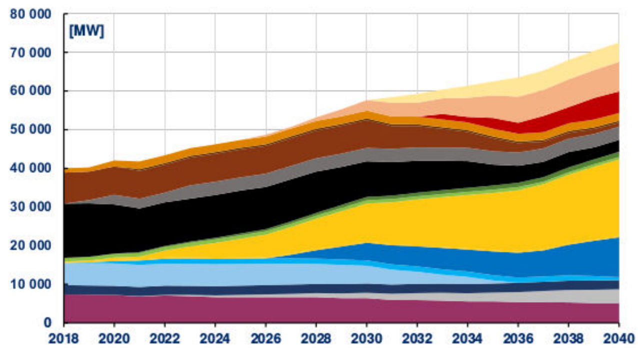

Figure 2. Installed net capacity by technology until 2040 [1]

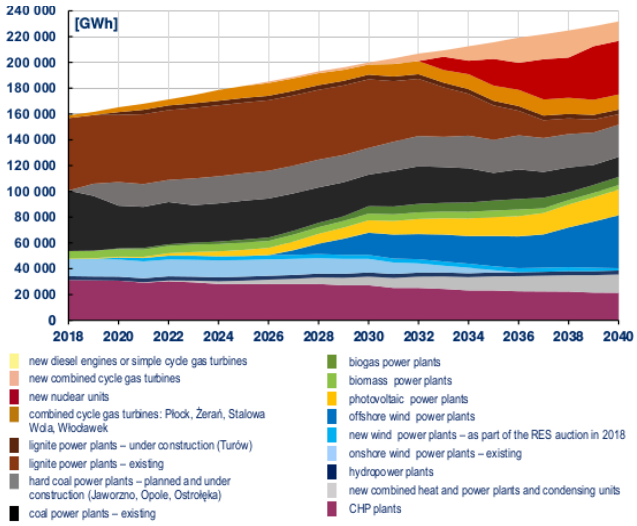

Figure 3. Electricity generation by technology until 2040 [1]

Locations taken into consideration for siting the first Polish NPP are Zarnowiec and Lubiatowo. To make it happen, PGE EJ1 and science communities are in the course of educating the public (especially from siting area) about nuclear power. Many local populations are supporting nulcear power plant because its attracts investments and creates numerous jobs onsite.

B. High Temperature Reactors

Draft of PEP2040 also mentions HTR as a potential heat source for industry. Nuclear power plants plans are shown on figure 6. Power and chemical companies use today coal- and gas-fired boilers to produce heat and most of them will need to be replaced in 2030-2050. In July 2016 Minister of Energy appointed special „Committee for deployment of high temperature reactors”. This committee published report in January 2018, pointing out thatlarge uncertainty on cost of CO2 emission makes investing in fossil fuels more risky, as opposed topotential reduction of CO2 emissions by 14-17 Mt/y (Poland hosts 13 chemical plants that need a combined value of 6500 MWt at 400-550°C) by utilizing nuclear power for industry. In National Centre for Nuclear Research (NCBJ) scientist and engineers are working to design and construct experimental HTGR to 2026.

There is also plan to work on designing a Dual Fluid Reactor (DFR). DFR is a new concept of nuclear reactor with two fluids: liquid lead as a coolant and molten salt as a fuel. This type of reactor can reach very high temperatures by using ceramics as construction materials.

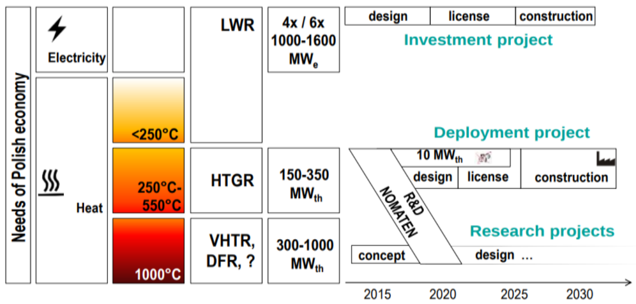

Figure 5. Nuclear roadmap of Poland [4]

C. Private Nuclear Power Plant

Michal Solowow, polish businessman and investor, one of the richest Poles and the owner of chemical company Synthos SA, is strongly interested in nuclear power plant for his factory. He said that small modular reactors can play a significant role in addressing Poland’s energy challenges, the modernization of the energy sector and in achieving necessary and responsible deep decarbonization. He also thinks that utilizing small modular reactors to generate clean energy will improve our chances to move away from coal and have a positive impact on our industry and nation [7].

On 21st of October 2019 Synthos SA and GE Hitachi Nuclear Energy signed Memorandum of Understanding [7]. This document is setting up a collaboration on potential deployment applications for BWRX-300 in Poland. BWRX-300 is 300 MWe water-cooled small module reactor with natural circulation and passive safety systems [8].

This information shows that in Poland there are private investors who are interested in nuclear power plant as his own energy source.

VII. CONCLUSION

Poland has tried to build a nuclear power plant for decades.

The first plans was in 1970s. It was planned to build a nuclear power plant in Zarnowiec and in five other sites. Unfortunately, despite the completion of 40% of the Zarnowiec NPP construction project has not been completed.Today, from the perspective of almost 30 years after abandoning the construction, black and white shows how bad a decision was to abandon the construction of the Żarnowiec Nuclear Power Plant. The really reason why polish NPP construction downfall is still not clear. The period when construction was abandoned falls on politically unstable times, there were ecological protests and financial problems.

The decision to discontinue the first Polish nuclear program meant that our generation sources are mainly coal sources. Therefore, Poland has significant CO2 emissions, which should be reduced in connection with the current climate policy of the European Union. Greater diversification of energy sources is therefore needed. For the country's energy security, the best option is a stable nuclear power plant. There seems to be no other option than to implement the current energy program to ensure energy security in the country. It looks like nuclear power in Poland is not only a possibility, but it is absolutely necessary.

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IYNC2020 SYDNEY

The environmental and social performance of nuclear reactor technology in Australia in comparison to other technologies

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I. Introduction

The anthropogenic influence on the climate is globally seen as one of the major threats to humanity. The environmental changes regarding weather patterns and global temperatures are driven by human activity. The industrial revolution sparked significant technological developments resulting in a substantial increase in emissions of greenhouse gases (GHG). The Intergovernmental Panel of Climate Change (IPCC) conducted research demonstrating the carbon dioxide concentration has increased from 275 parts per million to nearly 400 parts per million in less than 150 years [1]. These findings have sparked a catastrophic response from countries regarding their overall carbon footprint with the UK Parliament declaring a climate emergency and Labour Party leader Jeremy Corbyn announcing, “We have no time to waste”. This reinforces the significance of sustainable development and shifting to a low-carbon future.

The energy generation sector has a significant contribution to the GHG emissions of a country especially when fossil fuels are being used. The current Australian energy mix is dominated by coal and natural gas with approximately 94% of primary energy being produced by fossil fuels in 2017 [2]. This reinforces the need for low carbon technologies to be implemented in the Australian energy mix to reduce their carbon footprint. In order to reach the current target, set by the Paris Agreement, there needs to a significant change in the Australian energy sector.

The objective of this paper is to demonstrate the viability of nuclear power in Australia on an environmental and social basis. The environmental assessment involved conducting a life cycle assessment (LCA) assessing the overall footprint and identifying areas of improvement. Lastly, a survey was created to gauge the perception of Australians towards energy generation, the environment and nuclear technology.

II. Literature review

A. Environmental performance

Australia’s promise to reduce their emissions by 28% relative to 2005 levels by 2030 demonstrates the need for cleaner technologies [3]. The problem with renewable energy is the lack of energy capacity relative to fossil fuels showing this technology is unable to meet the current energy demand. Nuclear power offers an improved capacity combined with reliability and improved environmental performance. An initial comparison of life cycle assessments from different studies [3-22] demonstrate the variation in life cycle emissions due to different boundary conditions, reactor fuel and type. The Hondo (2005) study considered a Boiling Water Reactor (BWR) in Japan and sourced the uranium from Canada and Australia, however the type of uranium mine and ore grade was not specified. Alternatively, the Fthenakis and Kim (2007) study conducted multiple cases using different ore grades reinforcing the ore grade impact on the emissions during the uranium processing stages. Both studies agreed significant amount of life cycle emissions comes from uranium enrichment and preparation processes. This was supported by Siddiqui and Dincer (2017) reinforcing the large contribution of emissions from the mining and milling processes. Furthermore, the Lenzen (2008) review identified the effects of the differences between shale and ore mining, and wet and dry processes for uranium conversion.

There are other variables that impact the GHG emissions of this technology as shown through many studies. Sovacool (2008) demonstrated the effect the assumed reactor life has on the overall environmental impact with an increase in lifetime from 30 to 40 years can decrease the GHG emissions by 23%. Furthermore, this study pointed out the difference between heavy water and light water reactors hence, the selected reactor type will impact the LCA results. Another study conducted by the Department of the Prime Minister and Cabinet in 2006 considered a nuclear industry in Australia providing 10% of the energy demand [27]. The results reinforced the low-carbon nature of the technology and provided insight into an Australian strategy to reach their emission targets.

All these studies [3-15, 17-22] support each other in respect to the low environmental impact of nuclear technology. However, these are challenged by two physicists, Jan-Willem Storm van Leeuwen and Philip Smith who conducted an LCA [16] that estimated significantly higher GHG emissions. They assumed a higher energy requirement for the construction and decommissioning phases compared to other studies. Furthermore, they assumed the implementation of this technology would exhaust uranium reserves resulting in lower grade uranium thus, increasing the energy usage. The other studies do not consider the impact this technology may have on uranium mines and processing facilities. Leeuwen and Smith demonstrate the importance of considering this effect and how current studies may not model future conditions.

B. Social perception

Polling is a common technique used to gauge the attitude and perception of the public however, the validity of results needs to be considered by looking at: the organisation interests for specific outcomes, the wording or ordering of questions and the sample size [23]. Currently, a trend is evident between countries with nuclear capacity and the public support for the technology. The European Commission conducted a study demonstrating the effect climate change had on the public support for nuclear technology. The results demonstrated how environmental issues such as, climate change can encourage the public to support a technology even though they have other concerns.

A study conducted by Bird et al (2014) examined Australia's perception of nuclear technology with two surveys, pre (2010) and post (2012) the Fukushima disaster. The study looked at the effect of 'Reluctance Acceptance' that looks as the negotiation of a person's opinion on nuclear power due to their attitude towards climate change [24]. The results demonstrated the public's belief that coal is increasing climate change and causing significant air pollution. However, 81.9% of responses believed nuclear power creates dangerous waste hence should not be implemented [24]. Furthermore, the survey saw only 24.4% supporting nuclear power and 30.2% believing the benefits outweigh the risks [24]. These results demonstrate a social barrier for this technology to be implemented in Australia.

III. Methodology

A. Life cycle assessment

A life cycle assessment (LCA) is an accepted method for analyzing the environmental performance of specific technologies and identifies key hot spots in the life cycle. This is a time and resource intensive process with data collection creating problems due to limited data availability. It is important to note this approach does not assess the technical performance, cost, or political and social acceptance of a technology.

This LCA was conducted using GaBi software aiming to demonstrate the environmental performance of nuclear power using a functional unit of 1 MWh of electricity produced. The major impacts considered includes, global warming potential (GWP), acidification potential (AP), eutrophication potential (EP) and particulate matter (PM) emissions. A major outcome of this research was identifying there is a lack of available data making it difficult to quantify inputs and outputs. A life cycle inventory was sourced from the National Energy Technology Laboratory [25] and the database included in the GaBi LCA software. This software was used to carry out the LCA and provided specific inputs and outputs for transportation and the Australian energy mix. The boundaries of this assessment included, uranium mining/milling, conversion, enrichment, fuel fabrication, reactor construction, operation, decommissioning and transportation.

The selected technique for uranium mining and milling was open pit mining with Ranger and Olympic Dam mines using this method [26]. For the conversion stage, the selected technique was the wet process that is currently being used by the Cameco Facility in Ontario (85% fuel efficiency) [25]. The chosen method for the enrichment process was gas centrifuge (up to 5% weight) used by the URENCO facility in America [25]. The selected reactor type was limited to the data available from the inventory thus, this study considered a Gen III reactor with a lifetime of 60 years [25]. Efforts were made to assessing the performance of small modular reactors however, this technology is still novel with limited information available. This technology would be appropriate for Australia due to the modularity and factory-fabricated components, and the low population density however research and development is required.

The transportation stage included, road and sea transportation for uranium processing. This study considered uranium being mined and milled in Australia then transported to the USA for processing then transported back to Australia for use. The GaBi software provided inputs and outputs for road and sea transportation using Euro 4, 28-34 tonne gross trucks and container ships with 275,000 dead weight tonne respectively. These represent the European standards thus, it was assumed the inputs and outputs would be similar to Australian standards.

B. Public survey

A survey was conducted to gauge the public's perception on climate change, energy generation and nuclear technology. The questions were developed based on previous surveys conducted in Europe and created using Google Forms. The objective of this study was to assess the public's knowledge of current means of energy generation, the relevant environmental impacts, and how they believe we should approach this environmental challenge. The survey was advertised and handed out through social media and around Curtin University (Perth WA) between the 3rd of August 2018 and the 28th of September 2018. This method received 174 responses with a large proportion of student participation between the age bracket 18-29 (87.3%). This was expected as survey disposal was concentrated around the university.

IV. Results and Discussion

A. Environmental performance

The framework for this assessment was CML 2015 which provided a summary of the environmental impact of this technology. The GWP, AP, EP and PM values are shown in Table 1. In order to validate these results, the values were compared to literature LCA values as displayed in Figure 1.

Table 1: Summary of the LCA results regarding environmental impact

Indicator	Output
GWP	40.7 kg CO_2-eq/MWh
AP	0.178 kg SO_2-eq/MWh
EP	0.051 kg PO_4-eq/MWh
PM	0.029 particles/MWh

As seen in Figure 1, the emission values from this study are consistently greater than the literature average however, further analysis demonstrates these values fit within the range of the literature values. The higher values can be explained by considering the transportation that is required for uranium processing. Additionally, as discussed in the literature review, the selected reactor type, lifetime and method for each processing stage would significantly impact the results of the assessment. Furthermore, this study reinforced the large environmental impact contribution from the uranium processing stage as demonstrated by studies discussed in the literature review.

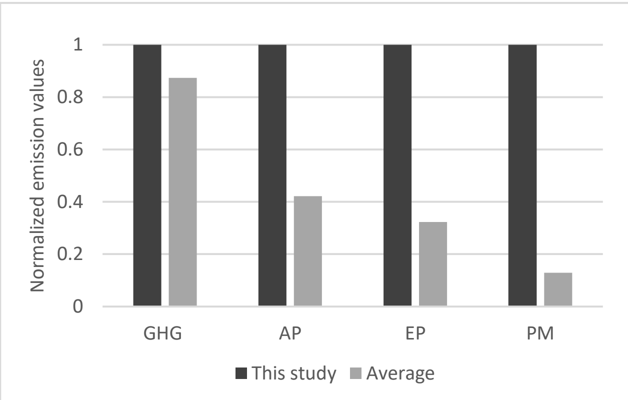

Figure 1: The comparison of the emissions from this study and the literature average

However, this study did approximate the impact of the construction, operation and decommissioning at a lower value compared to literature. Especially, the operation stage as this study considered a zero carbon dioxide output. The construction and decommissioning emissions were 31% and 80% lower than the literature average. However, studying the literature values in detail, the results from Leeuwen and Smith (2007) greatly over-estimated these values raising the calculated average. These outputs for construction and decommissioning from this study fit within the range of the literature values.

Lastly, these results were compared to other energy generation technologies to identify the capacity of nuclear power to tackle current environmental challenges. The GWP, AP and EP outputs for nuclear technology were similar to renewable energy from wind and solar, and significantly lower than coal and gas emissions as seen in Figure 2. On a life cycle basis, coal

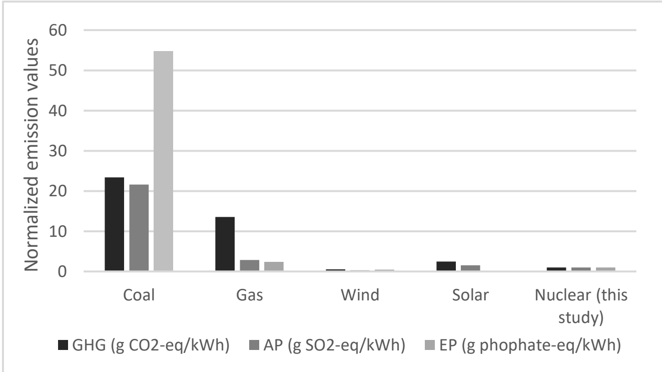

Figure 2: The comparison of life cycle GHG, AP, EP emissions across different energy generation technologies

emits 24, 22 and 54 times more carbon dioxide (GHG), sulfur dioxide (AP) and phosphate (EP) equivalents respectively compared to this study. Similarly, natural gas emits 14, 3 and 2 times more emissions in comparison. This reinforces the capability of nuclear technology being used to help Australia reach their emission targets.

B. Social performance

The first group of questions related to energy generation and the public's view on certain aspects of this sector. Firstly, it is important to gauge the public perception on climate change and whether they believe this an important challenge to tackle. Majority of responses (93.7%) agreed human activity is affecting/increasing the effects of climate change followed by 93.7% believing the public must take responsibility and 92.5% stating the Australian Government should invest in environmentally friendly technologies. This is reinforced when the participants were asked if they would consider paying more for electricity that is sustainably produced with 84.4% of responses willing to pay more.

Table 2: Summary of responses (%) to specific survey questions

Question	Agree (%)	Neutral (%)	Disagree (%)
Human activity increasing climate change	94	5	2
Public should act against climate change	94	6	0
Government must invest in clean energy	93	6	1
Nuclear will meet energy demand	58	33	9
Nuclear addresses climate change	56	33	11
Nuclear needs further development	60	22	18
Benefits do outweigh the risk	46	36	18
Nuclear is safe	42	35	23
Concerned about nuclear waste	77	0	23
Worried about radioactivity	60	18	22

The participants were asked what their most weighted consideration was when assessing energy technologies. There was a strong response (78.9%) selecting 'Climate Change solution' reinforcing the public's view towards tackling climate change. Additionally, 94% of participants believed fossil fuel technologies had the greatest environmental impact compared to nuclear and renewable energy. This demonstrates the public's knowledge of the downside of the current means of energy generation. Additionally, there were 83.9% of responses that believed renewable energy should be the centre of research and development with only 13.3% selecting nuclear energy.

The next group of questions related to nuclear power and assessed the public perception of this technology. Firstly, the survey asked participants to self-assess their knowledge of this technology resulting in an average score of 4.6 out of 10. This was compared to their self-assessed knowledge of current energy generation technologies which received an average score of 5.6. Furthermore, 65.7% of responses agreed nuclear power offered a clean alternative for power generation, and 56.1% believed nuclear power can address the current environmental challenges with 32.9% being unsure. Furthermore, a strong response (56.1%) believed this technology can meet the Australian energy demand however, 60.1% stated that nuclear power needs to be further developed before implementation in Australia. The public understands this technology is clean and viable but, the faith in safe operation is still missing creating a major barrier. Moreover, only 17.9% of responses believed the benefits do not outweigh the risks providing insight into how the public assess the risk and benefits of this technology. However, when asked about the safety of this technology, there were an even spread of responses. This reinforces the public perception of nuclear technology being unsafe, possibly due to past disasters such as Fukushima and Chernobyl. Additionally, 77.2% were concerned about nuclear waste disposal which is a current problem with this technology and 59.9% were worried about the radioactivity present in this technology.

The survey provided insight into the public's perspective on the operation and regulation of nuclear power plants. A low proportion of responses (38.7%) believed the Australian government will be able to regulate this technology while only 50.9% believed Australia could operate this technology without incident. This provides insight into the public's limited confidence in Australia's regulatory practices and demonstrates a major barrier for this technology.

Lastly, the survey included extended answers to allow participants to provide their opinion on energy generation. Common responses regarding the risks and problems associated with nuclear power included, waste disposal, accidents (e.g. nuclear meltdowns), radiation/radioactivity, human error and politics. Furthermore, there was a strong response (60%) believing nuclear energy has a place in Australia with only 20% believing otherwise. The common explanations for this positive attitude included, Australia's resource availability (land and uranium), another source of energy and offering a climate change solution. Alternatively, the responses believing there is no place stated the safety issues and economics of this technology, and the belief renewable energy should be encouraged instead. This attitude towards renewable energy was reinforced when participants were asked how Australia should tackle climate change which saw 70% of responses believing Australia should implement renewable technology. Other responses mentioned the importance of research and initiatives to encourage people to study alternatives and the education of the public on these environmental problems.

V. Conclusions and Further Work

The LCA reinforces the improved environmental performance of nuclear power compared to other energy generation technologies. This offers a viable solution to the current environmental challenges and emission targets. Australia is one of these countries promising to reduce their emissions however, they are still dominated by coal and gas technologies. There are current pushes to implement renewable energy however, with the current energy demand from the Australian population, these are unviable solutions. Nuclear technology offers a high energy capacity to meet this demand while improving the Australian energy generation sector environmental performance. This technology offers low GWP, AP, EP and PM emissions compared to fossil fuels considering the whole-of-life of each technology.

A major barrier for this technology is the public acceptance and attitude towards it. This study considered this and gauged the public perception of three different areas: environment, energy generation and nuclear technology. The key conclusions from this social study included, the public attitude towards environmental challenges and their belief that we must reduce our carbon footprint. Additionally, they understood the negative impacts of Australia's current means of energy generation and considers the environmental performance as a major priority when assessing technology. Furthermore, the public understood the improved environmental performance of nuclear power however, they are still concerned about the safety issues. But, majority of the participants believed nuclear power had a place in Australia due to Australia's natural resources and the reduced emissions.

Australia has committed to reducing their emissions by 2050 thus, significant changes are needed in the energy sector. The development of small modular reactors is still novel with limited information thus, further research and development is required. Additionally, the limited data available for nuclear power in regard to performing an LCA demonstrates a gap in research, and a reliable inventory should be created. As mentioned by the survey participants, the problems with nuclear waste handling, disposal and storage is a major barrier for this technology hence, it is important for further analysis into fuel cycles to reduce the amount of waste produced per cycle.

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Addresing Gender Imabalance: Context, Case and Considerations

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I. Introduction

Demographic imbalances (notably, but not limited to, gender imbalance) has been a long-standing issue in STEM fields, including the nuclear industry. The nature of this issues, and attempts to resolve it, are complex and require a collaborative approach. This paper provides a literature review that discusses the context of the problem, its significance, and a few considerations that should be made in the process of addressing the issue.

II. Context in STEM

One of the most universal challenges facing science, technology, engineering and mathematics (STEM) fields around the globe is demographic imbalances, including, notably, imbalance in gender distribution. A 2017 report by the Office of the Chief Scientist reported that women make up only 27% of the STEM workforce in Auistralia[1]. Not only is there an imbalance in total representation, but there is a significant imbalance in higher earning positions, reflected by the fact that only 12% of women in STEM fall in the top income bracket in Australia, compared to 32% of men[2].

The picture in Australia is not dissimilar to the worldwide picture; UNESCO's 2015 report found that under 29% of researchers are female[3]. There is significant variation in rates between regions, with some regions showing equal or higher percentages of women in STEM relative to men[3]. Interestingly, the relationship between women in STEM and gender equity in wider society is complicated, with some evidence showing that countries with higher imbalances in STEM were often those with greater gender equity in society more generally[4]. Further analysis in this paper suggested that quality of life pressures in countries with worse gender equity records may push women towards science. Given the complicated nature of this relationship, it can be suggested that it is important to consider the STEM-specific context of gender imbalances and initiatives to address these balances, and not only look at imbalances in a broader societal context. These imbalances in STEM more broadly trend in a similar manner in the nuclear industry[5].

While UNESCO indicates that trends in education are more promising, they note that a drop-off appears to occur at higher education levels[6]. While the number of women with tertiary qualifications in STEM is increasing (23% increase between 2006 and 2011), this number is in fact lower than the increase outside of STEM (31% in the same period) [2].

While this paper focuses on imbalances across gender (for the purposes of this paper, specifically focusing on male and female), it is worth noting that it is not the only demographic area in which STEM shows some imbalance from average population demographics. An Australian demographic report from the Office of the Chief Scientist in 2011[1] indicated that a higher than average percentage of those in STEM fields were born outside Australia (35% in STEM compared to 31% in non-STEM). Additionally, while science showed a similar age profile to non-STEM fields, engineering and mathematics both showed a lower than average proportion of people in earlier career phases (those under 35 accounting for 24% of the engineering workforce, 22% of the mathematics workforce and 33% of non-STEM workforces).

A number of reasons have been proposed to explain these differences. Professionals Australia's 2018 survey report[2] indicated a few possible reasons, including high attrition from the STEM workforce, policy towards maternity leave resulting in detrimental career outcomes, continued prevalence of gender-based harassment and discrimination, unconscious bias in recruiting and gender stereotyping. It is also worth noting that there are other cultural issues that may influence women considering STEM fields. For instance, this report also points towards lack of access to female role models in STEM, which can have a discouraging effect on both women in the workforce and younger women considering a career path in STEM. Current imbalances and historical trends can reinforce the image of STEM as a traditional male career.

III. The Case for Diversity

The case for diversity (or people, skills and thought) is broken here into three imperatives; ethical, safety and business.

Firstly, it is argued that aggressing gender inequity is ethically the right thing to do. While the ethical reasons behind addressing imbalances at work are ultimately subjective, it is reasonable to consider removing barriers that may block someone from a career path of their interest is a worthy cause[7]. The significance of the imbalance in STEM is large enough to indicate that there are systematic reasons for this imbalance (outlined earlier).

Secondly, a diverse workforce has benefits in the safety of a workplace. A diverse workforce and an inclusive workplace can improve workplace safety in two ways. Firstly, proponents of workplace health and safety have begun to incorporate psychological well-being as a component of workplace safety[8, 9]. Inclusive workplaces have been linked to lower rates of common indicators of well-being, including conflicts, grievances, turnover and absenteeism, all reflective measures of psychological well-being[10]. Secondly, increasing diversity of thought, and creating an inclusive environment in which people can speak up against common thoughts and practices, helps create an environment in which people are more likely to speak up when they see unsafe practices[10], a consideration that is highly important across many STEM fields.

Thirdly, a diverse workforce can also benefit organizations in a business sense. Workplaces that bring together a range of diverse perspectives and thoughts are more capable of thinking innovatively, solving problems and performing better on a range of measures of strategic success[11, 12]. Innovation and problem solving, of course, are both widely considered as key skills relevant to success in STEM fields. Additionally, inclusive environments have been tied to psychological safety, which is an important factor in fostering creativity and innovation[10, 13], along with increasing engagement[8]. When Harvard Business Review conducted interviews with 24 CEOs, it was noted that diversity is now widely accepted as essential for business performance[14]. This is supported by a range of literature indicating that inclusive workplaces are more competitive in the global marketplace[15].

IV. Considerations in Practical Applications Aimed at Addressing the Gender Imbalance

The current section reflects on some valuable considerations to be made in the application of initiatives to address gender diversity (and gender diversity more generally). The following four considerations are not meant to be exhaustive, but rather are meant to reflect some of the most significant reflections from reviews of past change initiatives.

Firstly, when implementing change to address diversity issues, it is important to track meaningful data to assist on the journey. In order to determine whether a program has had a meaningful impact, it is important to collect before and after data points to determine efficacy. The nature of the relevant data may change depending on the purpose of the initiative, but it is important to think broadly about all data sources that may be useful. For instance, while data such as recruitment and turnover data is useful in showing changes in the demographic distribution of a workforce, data that drills deeper into employee sentiment, conflict at work and other factors may also provide useful insight into the nature of the issue[16.] Planning and tracking a range of relevant data before and after implementation of programs is essential to determining efficacy of changes, their limitations, and key areas to focus on.

Secondly, when implementing significant change, it is important to acknowledge that change is typically a slow process. While we may refer to implementation of projects, cultural change is not achieved simply by implementing a solution; many stages are involved in the change journey, each of which may take a significant amount of time[17]. The journey to a diverse and inclusive workplace is no exception. Furthermore, the problem may be exacerbated by the context of larger societal trends. For instance, let's examine simply the number of women in the field. The process of improving total gender balance is made more challenging due to an overall shortage of women in STEM more broadly[6]. In addition, recruitment of more women into an organisation may be limited by turnover and organizational growth; there are only a limited number of positions available within an organisation. Recruiting at 50/50 would, ignoring other factors such as variations in turnover rates across genders, require the entire workforce to turnover before parity is achieved. However, areas with growing nuclear and/or STEM sectors may be advantaged as a growing industry alleviates some of the above problem[18]. In any case, those seeking to enact meaningful change must be aware that the process is not a brief one.

Another important consideration is the need to be aware of our own blind spots. Our adeptness in identifying issues varies person-to-person and issue-to-issue; for instance, evidence suggests that men often identify sexism towards women at lower rates than women identify sexism towards women[19]. In addition, we all to some extent show unconscious bias towards certain groups[7, 20]. Indeed, this is a good example in which a diverse team ensures that individual biases may be balanced our by differing perspectives amongst the team. The potential impact of our biases is also a great reason to ensure adequate data tracking, to determine any common trends in our practices or any areas of shortcoming that we may not personally identify. Being aware of our limitations, and ensuring we have teams and systems that can mitigate these limitations, is valuable in ensuring effective change.

While a complete picture of all practical considerations that must be made cannot be summed up simply, this paper will make one more point that is particularly worth taking into consideration; when addressing gender imbalances and diversity more generally, there is no optimal solution or procedure to follow. A US study of a range of corporations with diversity management programs found that many had not been successful in achieving significant change[21]. It is suggested that the most effective approach is to look at successful programs elsewhere to guide your approach[22]. In the nuclear industry, there are two key directions in which to look. Firstly, looking at successful programs in other nuclear organizations is a great way to find programs that may translate well to the context of another nuclear organisation. However, given that gender trends in nuclear are relatively reflective or STEM more broadly[23], and that gender trends do show some variations internationally[4], we should also seek to learn from other STEM organizations within our own nations. A combination of these two approaches may serve as the best way to inform an effective approach to improving the gender balance.

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A Tough and Bumpy Path: the Success of the First Pro-Nuclear Referendum in Taiwan

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I. INTRODUCTION

In November 2018, the "Nuclear-free Homeland in 2025" referendum, supporting the abolishment of nuclear power, was passed in Taiwan with 59% of the vote. This was the first time that the pro-nuclear voice had stood out since the nation first started using nuclear reactors to generate electricity in the 1970s.

Nuclear power generation in Taiwan faced numerous obstacles following the Chernobyl and Fukushima accidents. The current ruling party, which is often referred to as the anti-nuclear party, decided to amend the Electricity Act in January 2017 so that all nuclear energy based power-generating facilities have their operating licenses expire and cease operation by 2025. The expected share of electricity released by the government for 2025 is 50% natural gas, 30% coal and 20% renewables.

A massive blackout, which affected around seven million households, occurred six months after the amendment passed. Insufficient power supply in the nation started to attract more attention from the public. In January 2018, a referendum was proposed by the co-founder of a pro-nuclear group and two other scholars. The process of collecting signatures wasn't smooth in the beginning, but, in this case, we got to see how grassroots movements led to a consensus without support from any parties or organizations.

II. FROM ONE OF THE TEN MAJOR CONSTRUCTION PROJECTS IN THE 1970S TO A PLAGUE-LIKE ISSUE AND FINALLY BACK TO SUCCESS

A. *A History of Nuclear Power Generation in Taiwan*

There are eight commercial reactors located in four nuclear power plants in Taiwan, but two of the reactors have never been activated. The total capacity of the first three power plants is around 5,000 MW, which accounted for 11.4% of total power generation in 2018.

The construction of the Chinshan nuclear power plant, known as the First Nuclear Power Plant, was one of the Ten Major Construction Projects in Taiwan in the 1970s. The country still lacked infrastructure, and the oil crisis in 1973 had a significant impact on the economy. The pro-nuclear government decided to carry out these construction projects and added a second phase of another 12 major construction projects, which contained a plan for the construction of the second and third nuclear power plant. The construction of the two units at the forth one first began in 1999, but still remain incomplete. The project was first planned in 1980, but it was delayed several times because of political issues and the Chernobyl and Fukushima accidents.

After the Fukushima accident, like all other countries using nuclear power, the regulator carried out a safety review of all the reactors in Taiwan. The Lungmen project had already been delayed for years, and the Fukushima accident intensified the public's distrust of the project. Most Taiwanese trust Japanese goods, so after the accident they held sentiments like, "If the accident could happen in Japan, it is strongly possible for something similar to happen in Taiwan." Even the Taiwan Power Company (Taipower Co.), the state-owned and only electricity company in Taiwan, asked different organizations like the World Association of Nuclear Operators (WANO) and the European Nuclear Safety Regulators Group (ENSREG) to implement different safety check procedures or peer review. A majority of the public still favored scrapping the project. That was the peak of the anti-nuclear movement in Taiwan. Posters or handouts reading "No nukes" and/or "No more Fukushima" could been seen in places like cafes, shops and bookstores across the nation. This pressured the pro-nuclear government (former president Ying-Jeou Ma) into deciding to seal the power plant in 2014. This was the first and only time in which both parties reached a consensus on phasing out nuclear. The Premier said it was "... to leave a better choice (of energy) for the next generation." The construction of the first unit was already completed at that time. The second unit was 95% complete, but the construction needed to stop immediately.

In 2016, the opposition party, sometimes referred to as the anti-nuclear party, won the presidential election. The new president envisioned a nuclear phase out as one of her policies. She would realize this by not granting extensions to nuclear reactor licenses, investing largely in renewable energy and reducing the usage of coal. At the beginning of 2017, an amendment to the Electricity Act was passed - all nuclear power plants would cease operation by 2025.

TABLE I. COMMERCIAL NUCLEAR POWER PLANTS IN TAIWAN[1]

Nuclear Power Plant	*Net Capacity*	*Location*	*Status/ License Expiration*
Chinshan (1st NPP)	604MW x2	Northern Taiwan	Decommissioned Unit 1: December 2018 Unit 2: July 2019
Kuosheng (2nd NPP)	985MW x2	Northern Taiwan	Operational Unit 1: December 2021 Unit 2: March 2023
Maanshan (3rd NPP)	951MW x2	Southern Taiwan	Operational Unit 1: July 2024 Unit 2: May 2025
Lungmen (4th NPP)	1,350MW x2	Northeast Taiwan	Sealed Unit 1: Mothballed Unit 2: Construction suspended (95% complete)

A loose handle on one of the fuel assemblies was found during a refueling outage of the Chinshan nuclear power plant in December of 2014. This was the first time this incident had happened to a boiling water reactor (BWR) fuel assembly. Since then, the reactor has never resumed operation. Both units were decommissioned in December 2018 and July 2019 respectively.

In addition, the dry storage facility, which can hold fewer than 1,700 spent fuel assemblies, had been installed at the Chinshan nuclear power plant in 2014. So far, the municipality has not issued the permit for its soil and water conservation plan. For this reason, all the spent fuel will still be stored in the reactor temporarily before the dry storage facility starts operating. As it is located in the same city, the dry storage facility at the second nuclear power plant is in the same predicament as well.

B. The 729 Crisis of Insufficient Power Supply and the 815 Blackout in 2017

While both units at Chinshan went idle in 2014 and unit 2 at Kuosheng nuclear power plant entered the regular outage period in 2016, a coal-fired power plant in north Taiwan was damaged by a typhoon on July 29th 2017. Insufficient reserved capacity started to become a problem in the nation. Starting from July 29th, all government departments and most the state-owned businesses would have to turn off their air conditioners from 1 pm to 3 pm. This was during summer and was the hottest time of the year.

On August 15th 2017, valves of a gas supply pipe were turned off due to human error, which caused all of the units at Tatan, the biggest natural gas power plant in the nation, to shut down. A massive blackout happened at around 4:50 pm, affecting more than 6.5 million households in Taiwan. There are 6 units at the natural gas power plant, and its total capacity is around 4,500 MW, which means the whole nation was missing more than 10% of its electricity power generation in one second. Electricity rationing was implemented immediately, but the blackout still lasted for more than 6 hours in some areas. It was Taiwan's worst blackout in recent history.

According to both domestic and global media, these two issues caused more and more residents and business/company owners to contemplate whether Taiwan was too rushed in phasing out nuclear power plants. They even started to doubt whether the 2025 phase-out policy would be possible. Academics also offered officials their advice, a common thought being, "If all the nuclear reactors were operating when the accident happened at Tatan, we would be able to prevent this blackout." However, the ruling party still persists in proceeding with the phasing-out policy, and claims that there is a consensus for a "nuclear-free homeland".

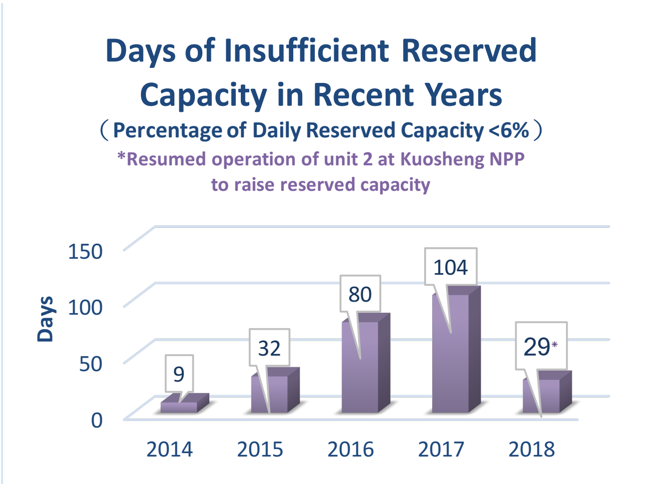

Figure 1. Days of Insufficient Reserved Capacity in Recent Years[2]

C. "Go Green with Nuclear" Referendum

On March 2nd 2018, the "Go Green with Nuclear" referendum was proposed jointly by three pro-nuclear advocates: Mr. Shih-Hsiu Huang, co-founder of the Nuclear Myth Busters, Professor Min Lee, Dean of the College of Nuclear Science of National Tsing Hua University (Taiwan), and Mr. Yen-Peng Liao, one of the Directors of the Chinese Society of Medical Physics (Taipei). The question asked, "Do you agree to repeal the first paragraph of Act 95 of the Electricity Act so that all nuclear based power-generating facilities shall cease operating by 2025?"

According to Professor Tsung-Kuang Yeh, Director of the Nuclear Science and Technology Development Center of National Tsing Hua University, we have two of the biggest coal fired and natural gas fired power plants in the world. Since the government began decreasing usage of nuclear power from 2015, more than 80% of Taiwan's power generation came from fossil fuel whereas renewables only comprised less than 5%.[3] If Taiwan are still to take nuclear power generation (comprising 9.3% in 2017 and then 11.4% in 2018 as the government decided to resume the operation of Unit 2 from Kuosheng NPP to fill the gap) out of the energy mix, it will definitely need to generate more energy from fossil fuels as its renewables are still inadequate to replace nuclear. Insufficient electricity would also threaten the manufacturing industry, which contributes to more than 30% of the gross domestic product (GDP) of the nation, including the world's largest semiconductor foundry - the Taiwan Semiconductor Manufacturing Company - and Foxconn, the world's largest provider of electronics manufacturing services.[4]

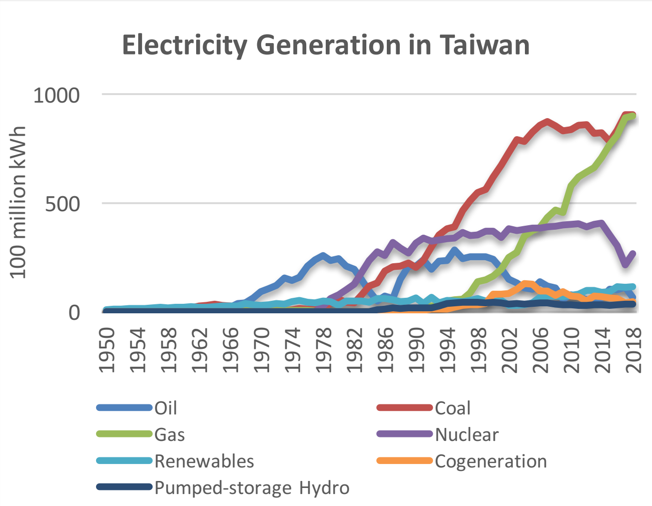

Figure 2. From 2014, coal and gas generation rises while nuclear generation falls[3]

Three thousand signatures had been collected for activating the proposal process of the referendum. After the application and the approval period was complete, it was already the end of June. In order to put the question alongside the local elections in November 2018, the Go Green with Nuclear crew would need to collect 300,000 signatures in two months. At the beginning of the endorsement period starting at the end of June, the information was only made available via the internet though the official Facebook page, some celebrities' and volunteers' social media and some faculty lectures. The number of endorsements started to increase but only at a very steady rate.

In order to collect enough signatures in support of the petition on time, the crew decided to take their message to the public in a more direct fashion. On August 25th 2018, the Surround Stations and Save Taiwan events happened at all the main train/metro stations in the nation. All the volunteers were divided to go to their assigned station to introduce Go Green with Nuclear to the public (with some debate) and attract the public to endorse it. Even former president Ma stepped out to endorse it. The experts from the crew also tried hard to gain exposure through the media. For example, they attended debate shows, provided commentary and advice etc. They also received a lot of support from friends overseas. Since then, the amount of endorsements started to raise dramatically, by around 20,000 per day, which was also a nightmare for the volunteers. All the endorsements were opened, sorted, bound and packed only by volunteers. Most of them were still college students or those who had just graduated from college, and the professors from the College of Nuclear Science of National Tsing Hua University.

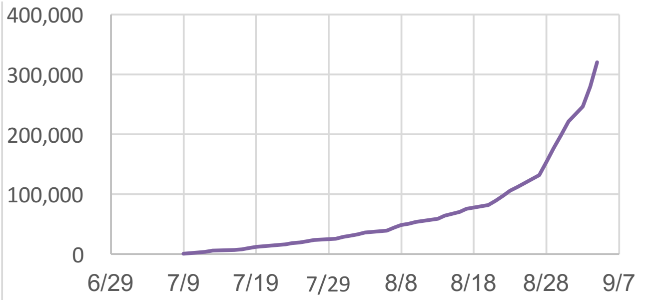

Figure 3. Endorsements received started to increase rapidly after the Surround Stations and Save Taiwan event on Aug 25th 2018[2]

On September 6th 2018, around 315,000 endorsements were packed in boxes and shipped to the Central Election Commission. However, after the validity check, only around 279,000 of them were valid, which was lower than the minimum required amount of 281,745. Another 24,000 endorsements were shipped to the commission on September 13th, one day before the deadline, but they were rejected. Mr. Shih-Hsiu Huang, one of the advocates who proposed the referendum, decided to go on a hunger strike immediately. He also filed a lawsuit against the commission after a few days. A month later, the High Court announced that the Central Election Commission must accept the second submission of the endorsements. The referendum was finally approved by the commission at the end of October to appear alongside the local election at the end of November. There was only one month left.

Figure 4. The volunteers who showed their support of Go Green with Nuclear at one of the busiest stations in Taipei City, Taiwan[2]

Despite there being a number of obstacles, pro-nuclear activists didn't waver in their goal of providing the public with information. They even did more after the referendum was allowed to be put alongside the local elections because there were still 4,939,267 votes needed to gain assent for the referendum.

Eventually, almost 60% of the voters casted in favor of repealing the nuclear phase-out policy. Interestingly, the number of voters who gave their assent were greater than the number of voters who gave their dissent in the cites/counties where nuclear power plants were located.

TABLE II. RESULT OF THE GO GREEN WITH NUCLEAR REFERENDUM

	Votes	%
Assent	5,895,560	59.49
Dissent	4,014,215	40.51
Valid	9,909,775	91.48
Invalid	922,960	8.52
Total	**10,832,735**	**100.00**

Source: Central Election Commission

D. *Reactions from the Government*

Around two weeks after the referendum was passed, the government announced that the nuclear-free homeland deadline (by 2025) on article 95-1 of the Electricity Act would be abolished. But the spokeswoman also said that the target of a nuclear-free homeland by 2025 of the administration still remains the same.

III. LESSONS LEARNED

In this case, we saw that going to the public to explain the reason we should maintain nuclear power in the country's energy mix was a lot more useful than being exposed to misinformation from those who have a bias against nuclear power. We also saw how the general public could reverse the president's policy without support from any party. Amending the law in this way was very meaningful to the public as it was the first time in Taiwan's history that this was done. The process of reversing nuclear policy was not easy, but it was ultimately successful.

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Case Study on the Nuclear Energy Supply Chain Efforts at the IAEA: How the Agency Identifies Priority Matters in the Nuclear Energy Industry and Develops a Program of Work

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I. Introduction

Over the past several years, the Nuclear Power Engineering Section (NPES), within the IAEA's Department of Nuclear Energy's Division of Nuclear Power, has developed a wide range of programmatic efforts and tools related to addressing the expressed needs of Member States on the matter on nuclear supply chain management. Within the context of this work, the nuclear supply chain involves all non-nuclear material and major components that are required to ensure safe, reliable operation of a nuclear power plant.

Over the past few decades the nuclear supply chain changed significantly . Increased globalization and a reduction in the number of traditional nuclear suppliers, makes the sourcing of nuclear components more complex. It is more difficult for operating organizations to locate suppliers for older, obsolete parts. The use of offshore suppliers requires long-distance auditing of supplier management systems and product quality, which can increase costs and the overall time required to obtain parts[1]. In some countries, over 20% of nuclear power plant equipment is obsolete [2]. Counterfeit and fraudulent items (CFIs) are of increasing concern in the nuclear industry and generally throughout the industrial and commercial supply chains [3].

In April 2018 the IAEA's Standing Advisory Group on Nuclear Energy (SAGNE) indicated that the largest threat to the world-wide nuclear industry was improper management of the supply chain, and recommended that the IAEA should "pursue wider international collaboration to manage and improve interfaces between regulators, technical support organizations, owner/operators and suppliers".

The objective of this presentation is to convey the inputs and checkpoints the IAEA balances when prioritizing a method of work and how the Agency subsequently develops and executes a program based on those priorities. Slow response to problems and lack of innovation are common criticisms of international organizations, but the Agency is actively working to innovate on the delivery of important content to Member States in a manner that is sustainable, substantive, and timely.

II. Brief History of IAEA Supply Chain Guidance

To date the most comprehensive guidance on the supply chain management elaborated by the IAEA is a document entitled "Procurement Engineering and Supply Chain Guidelines in Support of Operation and Maintenance of Nuclear Facilities" [2]. This document was published in the NE Series in 2016 and is both an update and expansion of the "Management of Procurement Activities in a Nuclear Installation" (published in 1997), particularly on the subject of supply chain organization activities. The publication provides a basic definition of supply chain activities, emphasizing that in addition to a conventional dimension of the supply chain management concept, "In the context of nuclear facilities, supply chain management implies an active role for the procurement and supply chain organizations within an operating organization, as opposed to a relatively passive role of simply issuing procurement specifications and responding to bids. It involves changing relationships and corresponding processes with external suppliers and within the operating organization itself". [4]

However, thorough review of this previous documentation does not reveal clear guidelines by the Agency in organization relating to the supply chain activities, especially in the light of interaction with the on-site contractors. So far, no guideline document with a specific focus on the supply chain management exists.

Despite no such specific documentation yet on supply chain management, some relevant references to the supply chain management methods or other related problems can be found in several publications issued within the NE and NS Series [5] [6] [7].

III. Developing Guidance Using a Global Perspective

The IAEA has several standard ways to obtain relevant information from Member States and organizations including: Consultancy Meetings, Technical Meetings, Workshops, Advisory Boards, Case Studies, Working Sessions and Expert Missions. The formality of each meeting varies, but their purposes remain the same—to gather relevant and diverse perspectives from around the world to solve common problems being faced by the nuclear industry at large.

A. Defining the challenge and outlining an objective

The major challenges and trends facing supply chain management in the nuclear industry, according to the Agency from utilizing the aforementioned methods can be summarized as follows:

1. The increased globalization of the nuclear supply chain greatly impacts countries embarking on new nuclear power programs, as well as those currently operating nuclear powerplants.

2. Operating organizations, vendors and suppliers must learn to adapt to the changing regulatory requirements and a global market place.

3. Establishing management systems, including comprehensive procurement and supply chain oversight processes with qualified personnel, is not an easy task.

Once these perspectives were identified, an agency objective to meet these challenges was established:

"*To provide information and guidance to Member States regarding good practices for management of procurement and supply chain activities related to the construction, operation and maintenance of nuclear facilities*".

B. Process for seeking budgetary support

The objectivewas clear and had support from a SAGNE recommendation. In addition, SAGNE also highlighted that the IAEA should develop "an online platform or portal for information and resource sharing, such as a Nuclear Supply Chain Toolkit". This recommendation was not made in a vacuum. SAGNE was also informed by the outcomes of two previous technical meetings where a similar recommendation was made by member states: Technical Meeting on Recent Developments in International and National Management System Standards, Including Quality Management Aspects (December 2017) and a Technical Meeting on Quality Assurance and Quality Control Activities as Part of a Nuclear Power Plants Management System: Lessons Learned and Good Practices (November 2018).

As a result of these recommendations, IAEA began a project, under the funding line of U.S. Peaceful Uses Initiative (PUI) extrabudgetary funds, to initiate preliminary efforts to meet the stated objective.

The IAEA has several budget lines to fund its programs: Regular Budget, the Technical Cooperation Fund, Extrabudgetary Contributions and PUI funds. PUI funds, a type of extrabudgetary funding source, was launched in 2010 to supplement extrabudgetary programs and the Technical Cooperation Fund to support technical cooperation project and other unfunded projects of the IAEA in the areas of peaceful application of nuclear technology. This funding stream allows the agency to be more flexible and quicker in developing and responding to shifting priorities of Member States—such as the development of the supply chain program in late 2018 after the SAGNE recommendation and Technical Meeting Recommendation earlier that year.

C. Establishing a proposal and defining outputs

Based off existing U.S. PUI funds at the IAEA, NPES proposed a program of work for 2018-2020, consisting of three major planned outputs:

1. The development of a new guidance document on supply chain qualification methods, with supporting web-based tools, including information on the supply of services.

2. Active participation by Member States at relevant technical meetings, workshops, and conferences.

3. The development of a new training course on supply chain oversight, including a module on the identification of technological challenges such as unidentified changes in the supply chain and CFI items

The web-based tools indicated here are referred to as "toolkits" within the IAEA. Within NPES, toolkit titled "The Nuclear Contracting Toolkit" has been released and is accessible to the public. The IAEA defines a toolkit as a community of practice website that supports all levels of specific activities, such as supply chain management for a nuclear power plant. They aim to facilitate good practices, consistency, and contribute to the long-term and safe nuclear powerplant operations.

The driving force behind the recent push for toolkits is the desire to be able to release information to Member States in a timelier and more approachable/accessible? manner. Publication at the IAEA is a notoriously lengthy process, sometimes taking more than 4 years between the start of writing to final release. Toolkits, usually accessed via an IAEA NUCLEUS account, allow special networks of professionals to develop, access and contribute to IAEA managed content in a near real time manner.

IV. STATUS OF WORK AND NEXT STEPS

The IAEA held a "Pilot Course on Nuclear Supply Chain Management and Procurement" in late 2019. The course was well received and will be offered annually to Member States in the future. Two new toolkits have been developed—one focusing specifically on the nuclear supply chain and another focusing on assisting Member States compare and contrast nuclear relevant regulations, legislations, and engineering standards of current operating or emerging countries. There are several publications under review and one technical document, slated for release in early 2020, focusing on the Quality Aspects and Quality Control in nuclear facilities.

ACKNOWLEDGMENT

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TRACK 9: COMMUNICATION, EDUCATION AND KNOWLEDGE MANAGEMENT

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THE INDIAN YOUNG GENERATION – GUARANTORS FOR THE FUTURE OF NUCLEAR INDUSTRY

D. AWASARE

INDIAN YOUNG GENERATION IN NUCLEAR (IYGN), INDIA

NUCLEAR IN THE NETHERLANDS NO LONGER THE ELEPHANT IN THE ROOM

MALOU STEGEHUIS

URENCO, THE NETHERLANDS

ASSESSING HARMFUL ALGAL BLOOM PHENOMENON IN THE PHILIPPINES: A NATIONAL APPROACH

M.L. RAÑADA-MESTIZO, C.O. MENDOZA, E.J. STA. MARIA, R.S.D. TABBADA

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PROMOTING THE BENEFICIAL AND PEACEFUL APPLICATION OF NUCLEAR SCIENCE AND TECHNOLOGY IN THE PHILIPPINES: PYGN SUCCESS STORIES

ANDREA LUZ G. NERY, JORGE R. SAHAGUN, JERALD B. BONGALOS, JOAN L. TUGO, RISSA JANE V. AMPER, HANS JOSHUA V. DANTES, MICHELLE P. ARISPE, ELIJAH GELEE Q. CAJIPE AND NIÑA GRACE S. PINEDA

PHILIPPINE NUCLEAR RESEARCH INSTITUTE, PHILIPPINES

THE EFFECTIVENESS OF A CARD GAME FOR TEACHING SECONDARY SCHOOL STUDENTS ABOUT CLIMATE CHANGE AND ENERGY GENERATION

Y. YU, B. SWEENEY AND M. TRENKEL-LOPEZ

NUCLEARGRADUATES, UNITED KINGDOM

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M. BINGHAM

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HUMAN RESOURCE DEVELOPMENT COSIDERATIONS AND CHALLENGES FOR NUCLEAR POWER PROGRAMME OF BANGLADESH

NILA RANI KUNDU
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THE *DOS AND DON'TS* OF SCIENCE COMMUNICATION

J. LACKENBY, K.H. SIZELAND
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ASSESSING THE IMPORTANCE OF PUBLIC AWARENESS AND EDUCATION ON NUCLEAR APPLICATIONS IN THE DIAGNOSIS AND REATMENT OF CANCER IN LESOTHO.

M.P. MAKUTUTSA
GOVERNMENT OF LESOTHO, DEPARTMENT OF ENERGY, LESOTHO

ROLE OF TRAINING INFRASTRUCTURE AND INTERNATIONAL COOPERATION IN INTERNATIONALIZATION OF NUCLEAR EDUCATION AND TRAINING

E. VARSEEV, A. ZHEREBILOVA, I. ANDRIUSHIN, M. TALABANOV, D. AGAFONOV
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OVERCOMING BARRIERS ON COMMUNICATION, EDUCATION AND KNOWLEDGE MANAGEMENT IN NUCLEAR TECHNOLOGY IN AFRICAN SOCIETIES

ALEXANDER PHIRI
CHAMBESHI WATER AND SANITATION COMPANY, ZAMBIA

VIRTUAL REALITY EMERGENCY RESPONSE TRAINING AT THE IAEA

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CRITICAL SUCCESS FACTORS FOR IMPLEMENTING OF KNOWLEDGE MANAGENENT IN NUCLEAR ORGANIZATIONS: A FUZZY APPROACH

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THE INFLUENCE OF NATIONAL NUCLEAR YOUTH COMMUNITY (KOMMUN) ON INDONESIAN YOUTH'S KNOWLEDGE ABOUT NUCLEAR SCIENCE AND TECHNOLOGY

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NEW COMMUNICATION CAMPAIGN BY THE SPANISH ENS-YGN: BASIC COURSE OF MEDICAL APPLICATIONS OF NUCLEAR TECHNOLOGY

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HOW THE NUCLEAR INDUSTRY OF HIGHLY ECONOMICALLY DEVELOPED COUNTRIES MUST PREPARE FOR THE SHIFTING PARADIGM IN ATTITUDE OF THE NEXT GENERATION WORKFORCE

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PROFESSIONAL PUBLIC ACCREDITATION OF EDUCATIONAL PROGRAMS IMPLEMENTED BY ROSATOM

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Rosatom Youth Council. Dirersity of development.

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I. Introduction

For several years Rosatom paying more and more attention in increasing participation the young generation in nuclear among. Based on the Rosatom Youth Council in a lot of companies a local Young Generation Network has been established.

Since September 2018 the Rosatom Youth Council has been working in the frame of a seven area that include all over Rosatom scope of activity. Nowadays, our highly qualified and motivated young people who have been working for a couple of years in the nuclear field and already took over a lot of knowledge and experiences, have to begin make own steps to involve young generation to solving problems the nuclear industry.

The paper will briefly range of activities and seven areas of the Rosatom Youth Council. A selection of them will be chosen to highlight our way for the future of nuclear energy in Rosatom, e.g. communication with the youth, knowledge-transfer, improvement of links between the youth and their employers.

The main purpose is to point out:

There is a young generation, who is ready to take over the knowledge and the responsibility for involving youth to future of nuclear industry.

II. History of the 'Rosatom Youth'

Young people in the Rosatom share the same dedication and passion for nuclear technology with previous generations. We have our own opinions; we must also perform our own active contribution! It is our own future and we have to make our steps! And we are already doing that...

First of all, 'we' it's more than 80 000 young people all over the Russia. And two years ago we made our first step and who have been united our young generation councils over the last two year.

Speaking of chance it means more than ever speaking about our own future, it was real challenge. Only two moments: 700 young people and 3 month…

III. Use a chance

It was a big problem for us. Just imagine: 30 young people from 8 cities, 3 different time zones, have to do something together. Units had experience in similar projects. The first 3 weeks we built communication and learned to work in this mode.

What experience can you get by doing it? First, use absolutely all the tools for communication and work planning. Do not look for the best apps, look for yours. Slack, Trello, Google Docs, WatsApp, ClickUp, Skype, IvaVks, etc. - among all these useful programs, you will definitely find the ones that your team needs. Secondly, try to study the experience of others people and try to generate more ideas; at the prototyping stage, you will certainly encounter difficulties. Thirdly, prototype all ideas, trying to go through the idea step by step you remove unsuccessful decisions. And most importantly - communicate with experts from other areas, it will help you get a fresh look at your idea.

Figure 1. Hash tag #MyRosatom, RYC2018.

The main step to be taken at the very beginning is to choose the main topic of the event. Without this step, you won't know whether you are moving towards the goal. We have chosen our theme - #MyRosatom. This simple hash tag united absolutely every young nuclear people.

Seven areas of congress - it was our second major step. What to talk about with young nuclear scientists? In what areas we need to work? How we can highlight the important? And here everything is simple. We took all the most important things that are discussed in the nuclear industry! These were:

- Collaboration

- Digitalization
- Science and new businesses
- NPP construction
- Development of the nuclear center
- Safety
- International networking

Working groups were established and detailed first goals were worked out. In order to check the results have been planned webinars and workshops. Not always everything was smooth, sometimes it was necessary to abandon the initial ideas and start all over again. Together with Academy of Rosatom we developed the concept of this event.

After all these works, within five days young people discussed problems, new ways of development, ideas, new technologies, development professional networking.

Working groups were established and detailed first goals were worked out. In order to check the results have been planned webinars and workshops. Not always everything was smooth, sometimes it was necessary to abandon the initial ideas and start all over again. Together with Academy of Rosatom we developed the concept of this event.

New contacts and relationships were grown up by building the Rosatom Youth Congress (RYC) and are still well established between their participants. But they are not automatically transmitted between young people; they must be constructed again and again and maintained. Personal contacts improve cooperation and are independent of where we are working: at the power plant, the utility or for the fuel vendor.

Figure 2. CEO Rosatom, Alexey Lihachev, RYC2018.

The whole young nuclear community began called Rosatom Youth. RYC was the starting point for young projects. There are presently challenged by a number of issues, which are of critical importance for youth and the future of the nuclear industry.

IV. LIFE AFTER RYC

Even more young people support the activities and the list of these activities is very long. Today the work is mainly focused on the field's communication with the youth, knowledge-transfer in all our active seven areas of congress: collaboration, digitalization, science and new businesses, NPP construction, development of the nuclear center, safety, international networking.

Young experts are presently have serious projects in each areas, some of them:

Collaboration. Project «Youth Day Information». It is an online platform where only young nuclear scientists choose the main topics. RYC within 90 minutes answers the most exciting questions: social area, development of professional skills, digital platforms, etc.

Digitalization. Project «Communication space». A IT platform uniting 365 enterprises of the industry for collaboration. The vector of work allows young nuclear scientists to be the first to test new digital products in the industry. Only new, only the first.

NPP construction. Project «Glossary of Rosatom». A Russian-English database of terms used in the construction of NPP.

Science and new businesses. Project «Rosatom Scientific Conference. Together». A conference gathering scientists from all over Russia. Together with industry leaders, RYC formed a delegation of 60 young scientists. During the week, the participants worked out the steps to develop new technologies in the industry.

The joint activities are the best way to secure our nuclear life, and a maximum number of projects over the long term, and retain all future opportunities.

For example the meeting of "Leaders of changes" where gathered more than 200 young people with their projects or RYC2019, where took part in about 400 young people, who make their first steps in nuclear industry. At this event they could find a like-minded people, exchange opinions at panel sessions and could get advices on self-development.

Figure 3. Workshop, RYC2019.

The joint activities also are the best chance for the young generation to work together with their all over the world colleagues and to guarantee the development of the nuclear industry.

CONCLUSIONS

The role of young professionals largely depends on the initiative they take.

The gained experience and knowledge help us to move forward and realize our new ideas. If you want to do something - do it. Failed - do it again. Find like-minded people and review their strategy and vision with them. Chat. There are a lot of cool experts around you and everyone has the right advice. But in order to get this advice, you need to take not only the first step, but also the second and third. You must value the time of experts and then you will get what you want.

And also we - the young members of the staff in the nuclear field - are going to work for more than 30 years! And we - the present Rosatom Young Generation - want to have opportunities to transfer once our knowledge and our experiences to a next Young Generation! Therefore:

Be visible, meet the people as well as participate in all available nuclear activities

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The Indian Young Generation – Guarantors for The Future of Nuclear Industry

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I. Introduction

The installed capacity of nuclear power in India is 3.22 percent of the total installed electrical capacity. Currently India has 21 nuclear reactors operating with installed capacity of 6780 MW. With growing energy demands, India has planned to build and operate 21 more reactors by 2031 with a total capacity of 15700 MW. The expansion of nuclear power will help India to meet its energy demand required to accelerate its Gross Domestic Products (GDP) growth rate. Moreover, this expansion will also create many job opportunities to the young generation. However, the public acceptance will be a big concern for India as there is already great misinformation about the beneficial uses of nuclear energy.

Education and communication are crucial to increase public knowledge, awareness and understanding of the peaceful applications of nuclear Technology. For the last few years, Young Generation has been attracting great interest from all over India across all fields (of what?). Contribution of youth in the Indian nuclear sector is also remarkable; highly qualified and motivated young people are working in the Indian nuclear sector. With the proposed expansion of Nuclear Power generation in India, the youth of India can play a vital role to communicate the benefits of nuclear technology to the general public.

This paper aims to provide an in depth study relating to the initiatives and actions taken to communicate with the general public and the role of the youth in communicating this nuclear message.

II. Indian Publics and Nuclear Issues

Traditionally the general public did not consider nuclear issues worthy of attention, except when stimulated by a specific event, such as the debate surrounding the indefinite extension of the Nuclear Non-Proliferation Treaty (NPT) in the mid-1990s or the Fukushima incident which happened recently in Japan. As per the conventional wisdom, when it comes to nuclear power, the public as a whole is largely supportive. However, this support is lacking in communities where nuclear power plants (NPP) are located, with negative sentiments spread across the state.

Despite support from local government, there still remains a number people in the vicinity of the proposed new Nuclear Power Plant that because of misconception regard nuclear power as a serious risk. Owing to vehement public opposition, nuclear projects have been halted, shifted or cancelled in places like Haripur (West Bengal), Bhoothathankettu and Peringome (Kerala), and in Mithivirdi (Gujarat). This is likely to escalate in future, as more reactors are in the pipeline.

The various manifestations of social opposition to, and changing dynamics in social acceptance of, nuclear energy in India are matters of serious introspection [1].

III. Indian Youth And Nuclear Energy

A. *The role of Youth*

The youth are the backbone of a society and hence they determine the future of any given society. This is because all other age groups; children, teenagers, middle aged and senior citizens all rely on the youth and expect a lot from them. Therefore, due to the high dependence on youth in the society, they have a role to play because the future of our families, communities and the country lies in their hands. Educated youths from India are playing a vital role in the nuclear sector by operating nuclear power plants around-the-clock and helping to meet the country's increasing energy demands. Around 35 percent of the population working in Indian nuclear power plants is youth.

Few young individuals working in the nuclear sector are engaged in public awareness activities through which they educate people across the country about the benefits of nuclear energy and try to clear their misinterpretations or misconceptions about nuclear power. However, there were limitations on these young dynamic future leaders to participate in such activities. Firstly, most of them work for state owned organizations and they do not always get clearance from their employers to participate in such activities. Furthermore, there was no platform on a national level like the International Youth Nuclear Congress (IYNC) which works at an international level, through which they can present their innovative ideas in Nuclear Technology or network with other stakeholders outside their organizations.

B. *Birth of Indian Young Generation in Nulear*

Inspired by the admirable work of IYNC worldwide, a number of like-minded youths, some working in Indian nuclear power sector and others having keen interest in nuclear energy, collaborated to form the Indian Young Generation in Nuclear (IYGN) society. These individuals recognized the need to unify the professional activities within the diverse fields of nuclear science and technology. IYGN was registered under the Societies Registration act on 30th November, 2018 and is the only registered Non-Profit Organization (NPO) in the nuclear sector dedicatedly working for young people in India. IYGN has provided a platform for Indian youths to share their ideas, innovations, research findings in different areas of nuclear energy sector including but not limited to, fuel cycle, nuclear waste re-processing and waste management, Nuclear Power Operation and maintenance, Environment and radiation monitoring.

IYGN provides opportunities for a young generation of nuclear enthusiasts to develop leadership and professional skills, create life-long connections, engage and inform the public, and inspire today's nuclear technology professionals to meet the challenges of 21st century.

C. Objectives of IYGN

IYGN has a clear mission which states "We shall work together for creating a significant approach of awareness to spread about nuclear science and technology to general public of India". IYGN has set following objectives:

1) *To provide a forum for the professional development of its membership;*

2) *To inform and educate the general population about nuclear science and technology; and*

3) *To contribute to the exchange of information among generations of nuclear professionals*

4) *To encourage the public understanding of nuclear science and engineering.*

5) *To foster closer professional and personal relations among the member.*

6) *To cooperate with other scientific and professional groups having similar objectives;*

7) *To engage in such other activities as may be appropriate for the fulfillment of the objectives of Corporation. and*

8) *To stimulate membership growth and encouraging participation at locations throughout India.*

D. Activities of IYGN

1) *Use of Social media:*-Social media has been ingrained into our society today to such extent that it is virtually impossible for people to take you seriously if you are not on any social media platform. Social media is the most adapted communiction tool of today's younger generation. Since it's birth IYGN perfectly utilized social media such as Facebook, Twitter and LinkdIn, to connect with it's stakeholders in nuclear sector.

Within a span of 1 year, the followers of IYGN Facebook page escalated up to 13,872 and is till increasing day by day, which is considerably high as compared to other Facebook pages of NGOs in India. IYGN Facebook page is followed by different age groups and genders. As shown in Fig. 1, around 86 percentage of total followers are Men whereas, 13 percentage are Women followers. It can also be seen from this graph that around 52 percentage of the total followers, both Men and women, are from 18-24 age group followed by 25-34 age group which is around 22 percentage. Both of these age groups are considered as young age groups.

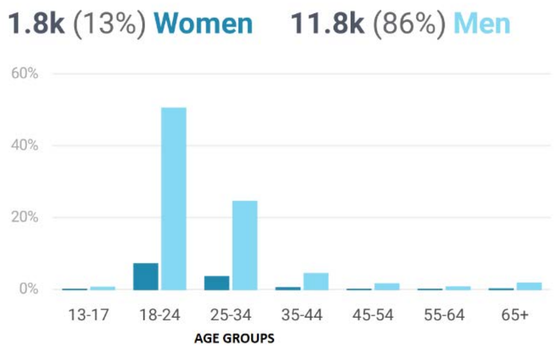

Figure 1. Age group wise and gender wise distribution of IYGN Facebook page followers.

IYGN also has followers, around 3 percentage, from 65+ age category which are mostly retired scientists and officers from India's Department of Atomic Energy (DAE) who constantly provide their valuable inputs and guide our younger community on Nuclear Energy related topics online.

Youngsters working for IYGN continuously keep adding new material such as important news from nuclear sector worldwide, job opportunities for young generation, career guidance, online quiz competition on nuclear energy, essay writing competitions, nuclear awareness videos and so on. This material is extremely helpful for young students and professionals interested in nuclear sector.

IYGN has also developed their state of the art website and is being visited by many people to seek more information about nuclear energy. IYGN members try to clear different doubts raised by their audience through online chat or messaging services of different social media platforms.

Figure 2. Glimpes of IYGN on field activities-distribution of fact sheets to students.

2) On field activities.-Apart from social networking activities, IYGN is also proactivley involded in public awareness programs where they meet directly with people, students and try to clear their mis-understandings about nuclear energy through presentations, displaying posters and short videos and distributing fact sheets (Pamphlets). Glimpes of interaction with students and distribution of Pamphlets by IYGN members are shown in fig.2 on previous page.

IYGN also arranges industrial visits to nuclear facilities in close association training management there where participants get their knowledge enriched. These visits let student to know things practically through interaction, working methods and employment practices in nuclear facility. Moreover, they give exposure from academic point of view. Main aim of these industrial visit is to provide an exposure to students about practical working environment of nuclear installations.

Figure 3. Youngsters participating in Nuclear for climate campaign.

3) Nuclear for climate campaign:- The Paris Agreement, signed in 2016 by 175 countries within the United Nations Framework Convention on Climate Change, calls to limit the increase in global average temperature to well below two degrees Celsius above pre-industrial levels. The Agreement points to a continued increased role in the use of nuclear power in the longer term. Its advantages in terms of climate change mitigation, as well as energy security and non-climatic environmental and socio-economic benefits, are important reasons why many countries intend to introduce nuclear power in the coming decades, or to expand existing programmes.

India has declared that nuclear energy is vital for meeting the challenge of climate change and suggested supporting efforts to promote its public acceptance. Indian young generation is playing significant role to promote Nuclear for climate campaign by arranging guest lectures on nuclear energy-green energy topic in different schools and colleges, displaying banners at roadside and asking nationals to submit selfies with slogans on nuclear for climate theme.

IV. Conclusion

For energy starving nations like India, Nuclear power remains an important option to meet the challenges of increased energy demand, address concerns about climate change, redress volatile fossil fuel prices and ensure security of the energy. Public acceptance is one of the key concerns while installing new nuclear reactor in India. Indian Young Generation plays significant role in endorsing nuclear energy and supporting nuclear for climate change campaign though several awareness activities conducted through social media and directly on field.

Acknowledgment

I am especially indebted to Mr. Pushya Mitra Singamaneni, secretary of Indian Young Generation in Nuclear society who have been supportive of my work and who contributed his valuable inputs to write this paper.

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Nuclear in the Netherlands
no longer the elephant in the room

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I. Introduction

The Dutch nuclear industry is very versatile and has a highly international outlook. What characterizes the Dutch industry is the presence of a complete nuclear supply chain. From the production of nuclear fuel via research and education to the production of medical isotopes and electricity ending with storage and management of radioactive waste. The sector supplies knowledge, services and products that are vital to society. But how exactly does that society think about nuclear power? A recent study by van Vliet showed that 54% is in favour of using nuclear energy [1]. But a report on the public opinion on nuclear power from the Social and Plan agency shows that nuclear power was never really popular with all time low in the second half of the eighties [2]. That raises the following question:

'How did nuclear energy lose its taboo in the Netherlands and what was Urenco' s share in that?'

II. Methodology

This paper gives an overview of activities that in some way have contributed to the sway in public opinion on nuclear in the Netherlands. Experience from working at Urenco combined with data from different public sources are the foundation of this paper.

III. Research

A. Background

Nuclear energy has, also in the Netherlands, a long history of high expectations, international research, political debates, affairs, (successful) research and protests. A very short outline shows that the first and only pressured water reactor was commissioned in 1973 [3]. That is the same year Urenco started its enrichment activities in Almelo [4]. On March 4, 1978 the biggest protest against nuclear energy took place in Almelo. More than 40.000 people came to Almelo to protest against nuclear arms and they combined this with a call against the expansion of Urenco and nuclear power [5]. In the years that followed, the sentiment changed gradually. There were no more large protests and even a new nuclear power plant was being discussed. Then Fukushima happened and nuclear power was back in the damn corner. This lasted until Sunday November 4th 2018, when Arjan Lubach [6], dedicated his show 'Zondag met Lubach' on getting nuclear out of the taboo sphere. It was thanks to this item that the discussion about nuclear energy, a subject on which there has been a taboo for decades, has been reopened [7]. Almost no one had expected such a revolutionary initiative from the progressive channel VPRO that gave Lubach, against the predominantly left-wing oppression, the opportunity to defend his divergent position. With credible arguments, Lubach has demonstrated that nuclear energy, as an alternative to wind and solar energy, is actually not such a bad idea at all. Even if the total surface area of the Netherlands were to be covered by wind turbines and solar cells, that would be far from enough to meet our energy needs at national level. The item was viewed by more than 1 million people and even led to parliamentary questions. So the contrast between 1978 and 2019 couldn't be bigger.

B. Urenco

Urenco provides uranium enrichment services and fuel cycle products for power generation within a framework of high environmental, social responsibility and corporate governance standards. From a communications perspective, the strategy has changed in 2014. Urenco believes that nuclear is key in achieving climate goals and that nuclear should be part of the energy mix. Below an explanation will be given on the efforts that Urenco has made to contribute on the changed public opinion.

C. Strategy

A two-track strategy was developed with focus on ambassadorship and political influence. Ambassadorship was chosen because change always starts with yourself. If the public opinion needed to change, Urenco had to lead by example. A good supply of background information and facts is key in creating more understanding and positive awareness around nuclear energy. Even though Urenco is a relatively small company that doesn't have the same influence as for example Greenpeace; a difference still can be made.

D. Ambassadors

In 2014 the first public campaign about powerful facts was published. The results were overwhelming: an increase of 7000% of website traffic, 20.8% increase in LinkedIn followers, 6% on Twitter and 12% on Facebook [8]. From the increased attention by the public could be learned that the public was eager to learn more about nuclear. The success of the campaign can be found in the bottom up strategy for engagement. A competition was organized amongst employees to come up with their own powerful facts. That first competition was the start of an ambassador engagement strategy where employees actively are encouraged to spread the message. The highlight (for now) of this strategy was the participation of 50 Urenco employees in the Nuclear Pride Fest in Munched. Organized by the Nuclear Pride Coalition in 2018.

Between 2014 and 2018 several campaigns have been conducted and the educational programs have expanded. In partnership with the BSA, a Richie inspired program of CREST classroom resources was developed. The CREST Star (aimed at 7-11year olds) and Discovery Award (aimed at 11-14 year olds) activities aim to inspire students to study STEM and provide further insight into the principles behind nuclear energy [9]. Next to that there are colleagues who proudly cooperate with online and offline media to talk about their work and their ambassadorship for nuclear.

E. Public affairs

The other part of the two track strategy was focused on public affairs and enlarging the political influence. As said before, the Dutch nuclear industry is very versatile and represents a complete nuclear supply chain. All six companies are united in Nuclear Netherlands [10] and working together to increase the political influence seemed the most logical thing to do. In 2014 the highly visible Nuclear Industry Summit was hosted, as a side event to the Nuclear Security Summit in the Hague. In 2017 the platform daytoday.nu was launched. An innovative platform with news and background stories about the international nuclear industry. From the people working in the industry to in depth articles on the developments around small modular reactors. This met a direct requirement of political stakeholders and the platform is now being embedded in the Nuclear Netherlands website. Focusing on public affairs for Urenco also means tightly monitoring of media. Mid 2019 an article was flagged that was published in national media stating that nuclear power plants have never been profitable [11]. It was based on a German study and widely spread and picked up by other media. Parliamentary questions we're asked. As a response, a substantiated statement was written, undersigned by Nuclear Netherlands, in which the facts from the article are refuted based on evidence. It is exactly this statement that was used to answer the parliamentary questions about cost of nuclear [12].

In addition, Urenco has proactively reached out to ministers and councilors to get them over to Almelo for a site tour with the aim to provide them with the correct background information which they could use in debates.

IV. Conclusion

But what is the true effect of all these (joint) efforts? In the beginning was already said that recent studies show a slight majority of the general public being in favor of nuclear energy. More recent online polls show an even greater percentage of people in favor. In October 2019, a poll from Dutch national paper Telegraaf showed that of the more that 7200 voters, 91% thinks that the Netherlands should seriously consider the construction of new nuclear power plants. 88% thinks that nuclear power is indispensable in achieving climate goals. [13]. In conclusion, the role of Urenco and Nuclear Netherlands in the political and public sway may not be 100% measurable, but the contribution to being no longer considered as the elephant in the room was definitely significant.

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Assessing harmful algal bloom phenomenon in the Philippines: a national approach

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I. INTRODUCTION

Studies on quantifying harmful algal toxins had significantly increased in the recent years in response to a lacking universal method. With the collaboration of international institutions (IAEA, WHO, NOAA), a radioligand receptor binding assay (RBA) had been established to quantify saxitoxins (Van Dolah et. al., 2012; 2009), and is being developed and adopted for ciguatoxins (Diaz-Asencio, et. al., 2018; Vacarizas, et. al., 2018; Bottein and Clausing, 2017) following the same experimental format. These methods are utilized in the Philippine setting to supplement and make effective a seafood monitoring program, thus safeguarding our national consumption of fish products.

Seafood products are considered to be a staple food by many Filipinos and are an essential economic commodity. According to the Bureau of Fisheries and Aquatic Resources or BFAR (2016), the country is among the top 10 fish-producing countries worldwide; summarized the fish products in Table I. And in 2018, the foreign trade performance of the fishery industry is at a net surplus of 911 million USD, 78% of its fish production includes fish, crustaceans, molluscs, and aquatic plants (including seaweeds).

TABLE I. PHILIPPINE EXPORT OF FISH AND FISHERY PRODUCTS BY KIND, QUANTITY AND VALUE, BFAR, 2016*

	Commodity/Kind	Quantity (MT)	FOB Value	
			('000 USD)	*('000 PHP)*
A.	Fish Crustaceans, Mollusks, etc. and Preparation	204,999	724,553	34,590,159
A.1	Fish fresh (live/dead) Chilled/Frozen	79,990	280,972	13,413,589
A.2	Fish, dried, salted/in brine; smoked fish	4,289	18,920	903,240
A.3	Crustaceans, mollusks and aquatic invertebrates	28,160	155,175	7,408,072
A.4	Fish and other aquatic invertebrate, prepared/preserved	92,560	269,486	12,865,259
B.	Shells and By-Products	4,041	19,279	920,369
C.	Miscellaneous fishery products and other by-products fishery products	48,179	206,604	9,863,257
	Grand Total	257,219	950,435	45,373,785

**https://www.bfar.da.gov.ph/profile?id=7*

Seafood, once contaminated with natural toxins, is a significant public health, economic and environmental issue world-wide. Understood as the Harmful Algal Bloom (HAB) phenomenon, the human health impact occurs when toxic phytoplankton are filtered from the water as food by shellfish and other carnivorous gastropods, and fishes (Shumway et al., 1990). These can rapidly accumulate within their system the algal toxins to levels which can be lethal to humans and other consumers (Sombrito, et. al., 2007; Van Dolah, 2000). These blooms also cause loss in the aquaculture industry, exceeding $1 million per occurrence (Corrales and MacLean, 1995). Over the years, there has been an increase in the incident of HAB leading to food-borne illness and deaths (Furio and Gonzales, 2002), loss of livelihood and alteration of the marine ecosystem, which makes a national government restoration effort highly costly.

II. ASSESSMENT OF A RISK MANAGEMENT PROGRAM

In the Philippines, paralytic shellfish poisoning (PSP) and Ciguatera poisoning (CP) are the most common HAB phenomena, where the former is mostly reported (Ching, et. al., 2015; Mendoza, et. al., 2013). The intermittent reporting of these cases indicates the lack of available efficient method on detecting such cases. Collaborating with the HAB monitoring agency of the country (Bureau of Fisheries and Aquatic Resources- Department of Agriculture, DA-BFAR), state university (Marine Science Institute, University of the Philippines, UP-MSI) and the International Atomic Energy Agency (IAEA), the Philippine Nuclear Research Institute (PNRI) initiated steps to undertake towards ultimate acceptance of RBA (Fig. 1).

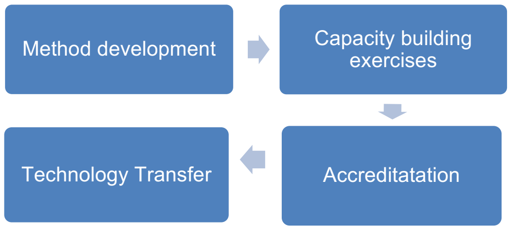

Figure 1. Steps in undertaking ultimate national acceptance of RBA for HAB monitoring application.

The country is still using the Mouse Bioassay (MBA) until the RBA for paralytic shellfish poisons was accredited as an Official Method of Analysis by AOAC International. Development of the method was conducted by NOAA, and interlaboratory comparison exercise were participated by both DOST-PNRI and DA-BFAR to achieve capabilities in conducting RBA for PSP for its subsequent accreditation (Van Dolah et. al., 2012; 2009). The same framework is also being utilized for the method development of RBA for Ciguatera poisoning, with the addition of adopting an efficient sampling program (Reguera, et. al., 2016), and development of a surveillance program spearheaded by the Epidemiological Bureau of the Department of Health (DOH-EB). With an efficient method, this contributes towards effective policymaking to reduce public health risks and economic hazards caused by the HAB.

III. Development of Validation of the Radioanalytical Techniques to Address Harmful Algal Toxins

Nuclear applications on different environmental studies had been developed in the past decades, showing applicability and efficiency. Among which utilizes RBA (Fig. 2) for quantifying harmful algal toxins. RBA characterizes the sensitivity and selectivity of the algal toxins towards receptor molecules through competition binding using radiolabeled toxins (Ruberu, et. al., 2017). As described by Van Dolah (2009) for paralytic shellfish toxins, the labelled saxitoxin, $[^3H]$-STX, competes with unlabeled STX or its derivatives to bind themselves to site 1 of the voltage-gated sodium channel (Na_v) receptor. Bound $[^3H]$-STX unto the receptor is measured using the liquid scintillation counter upon establishing binding equilibrium after one (1) hour. Toxicity is reported as total STX equivalence (STXeq). A conversion from concentration units to toxicity value is made to evaluate total toxicity also reported in different Philippine sampling sites (Rañada, et. al., 2016; Sombrito, et. al., 2007).

Figure 2. Major steps (from left to right) in conducting Receptor Binding Assay for HAB toxins: sample extraction, RBA proper, preparation for liquid scintillation counting.

The biokinetic assay format describes RBA effectively, in which the method can suitably be adopted for environmental monitoring. Utilizing the same concept with the RBA for STX, the RBA for ciguatoxins is being validated (Diaz-Asencio, et. al., 2018; Bottein and Clausing, 2017). A tritium-labeled brevetoxin ($[^3H]$-PbTx-3) is used in this format. Both brevetoxins and ciguatoxins bind with site 5 of the Na_v, with a higher potency of the latter; toxicity equivalence is then evaluated. The method is being verified using different Philippine fish samples.

IV. Public Awareness Program

One of the objectives of the project is to transfer the information to the public. Different information dissemination formats regarding HAB and RBA were created: in the form of a brochure (Fig. 3), which was distributed to several IAEA Members States of the Asia-Pacific region and the audience in several conference, and of a learning module, and through poster and oral presentations in scientific conferences.

The RBA brochure presents the problem of PSP in global scale, since the group likewise targets international audience. It provides information on the harmful effects of PSP in tourism and export industries, as well as its lethal effect in humans. Furthermore, it promotes the application of the use of the radio-ligand receptor assay to more effectively address the issue in comparison with MBA, the assay currently used.

Figure 3. RBA Brochure (2014)

A learning module composed of seven (7) videos about HAB and RBA was documented. The content was based on literatures and in-house documents of the PNRI HAB research team. It is posted in the PNRI website since March 29, 2017. (http://pnri.dost.gov.ph/index.php/pnri-publications/popular-video)

In addition to the above-mentioned information materials, several scientific presentations were delivered during the International Conference on Harmful Algae (ICHA) since 2014 showcasing the method development and validation of RBA, and its applications on seafood safety.

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Promoting the Beneficial and Peaceful Application of Nuclear Science and Technology in the Philippines: PYGN Success Stories

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I. Introduction

The Philippine Young Generation in Nuclear (PYGN) and the Philippine Nuclear Research Institute – Nuclear Information and Documentation Section (PNRI-NIDS) worked together to organize events and contests aimed to promote the beneficial and peaceful applications of nuclear science and technology to the Filipino youth. PYGN and PNRI-NIDS organized the 4th Philippine Nuclear Youth Summit (PNYS) that held last December 4, 2018, which was attended by 170 participants composed of student leaders and young professionals who gathered and participated in various activities: the Nuclear 101 Awareness Seminar, Technical Sessions, Research Proposal Writeshop, and Nuclear Amazing Race. The PYGN and PNRI-NIDS also organized the Nuclear Youth Forum for members of the PYGN Core Team to gather in order to learn more about debunking myths on radiation and radioactivity and the research projects being performed in the PNRI for them to share to their organizations and communities. Video-Making and Jingle-Making Contests were also organized to reach senior high school and college students around the country to make videos on nuclear science and technology that were shared on social media.

II. Materials and Methods

The results and highlights of the PYGN projects, in cooperation with the PNRI-NIDS, aimed to promote the beneficial and peaceful uses of nuclear science and technology to the Filipino youth are presented in this paper. Different ways of stimulating the interest of the young generation, such as social media and video-making contests, are also presented.

The Philippine Nuclear Youth Summit and the Nuclear Youth Forum were evaluated by the participants through evaluation forms provided. These evaluation forms asked questions on their reason for joining the events, the relevance of the activities and the topics of the talks, the clarity and effectiveness of the activities or speakers in sharing the benefits of nuclear science and technology, and the general organization of the event by the project teams. Participants were also asked to write their comments on the event in order to help improve the implementation of the succeeding events. Evaluation scores were categorized according to the following rating standards:

For the PNYS Video Making Contest, individuals or teams of up to three members were tasked to make 1-2 minute videos on the theme "Nuclear Science and Technology Working for You", where the entries must highlight the beneficial applications of nuclear science and technology to Filipinos, especially the young generation. The PNYS Video Making Contest received a total of 135 video entries. Entries were judged based on the following criteria: Relevance (30%), Content (30%), Impact and Creativity (20%) and Popularity (20%). Meanwhile, for the Jingle & MTV-making Contest, individuals or teams of up to seven members were tasked to submit 2- to 3-minute songs and music videos that highlight the beneficial and peaceful applications of nuclear science and technology. The Jingle & MTV-making contest received 29 songs and music video entries. Entries were judged based on the following criteria: Content (25%), Originality of Lyrics (15%), Originality of Music (15%), Impact and Creativity (25%) and Popularity (20%). For both contests, the popularity criterion was represented through the number of likes/reactions, number of shares, and the reach of each official entry post on the PNRI Facebook page.

TABLE I. Rating Standards for Evaluation Forms

Score	*Weighted Score*	*Descriptive Rating*
1.00	1.00 – 1.50	Poor
2.00	1.51 – 2.50	Fair
3.00	2.51 – 3.50	Satisfactory
4.00	3.51 – 4.50	Very Satisfactory
5.00	4.51 – 5.00	Excellent

III. Results and Discussion

A. *Evaluation of the Philippine Nuclear Youth Summit*

A total of 112 completed evaluation forms were collected form the participants. From these completed evaluation forms, we were able to get information on majority of the participants' ages and occupations, as well as their answers to the questions on the evaluation form about the relevance, effectiveness and organization of the event. The breakdown of the participants during PNYS 2018 is shown in Figure 1.

About 70% of the participants of the PNYS 2018 were students. This was seen as an advantage as the students chosen as participants were student leaders, thus they were pinpointed for possible partnerships with their clubs or organizations. Majority of the young professionals present were teachers, which was also seen as an advantage as they could pass the information they learned about the beneficial applications of nuclear science and technology to their students who were not able to attend the event.

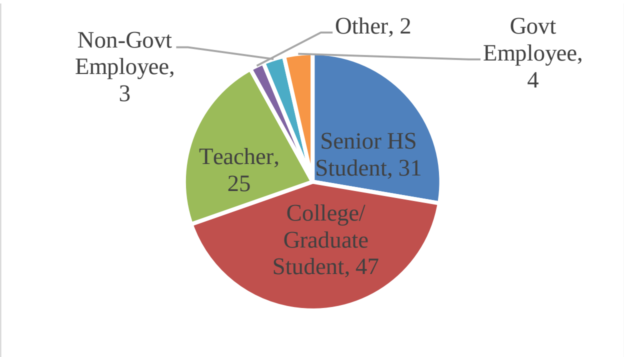

Figure 1. Number of PNYS 2018 Participants per Classification

Figure 2. Group photo taken during PNYS 2018

TABLE II. Mean Scores of PNYS 2018 Evaluations

Item on PNYS 2018 Evaluation Sheet	*Mean Score*	*Descriptive Rating*
Relevance of Nuclear 101 Talk: Overview of Radiation & Radioactivity	3.82	Very Satisfactory
Relevance of Nuclear 101 Talk: PNRI Services	3.88	Very Satisfactory
Relevance of Amazing Race	3.83	Very Satisfactory
Relevance of Research Proposal Writeshop	3.60	Very Satisfactory
Relevance of Technical Sessions	3.85	Very Satisfactory
The speakers were able to clearly and effectively communicate the messages to the audience. Do you agree or disagree?	3.88	Very Satisfactory
The topics were adequately explained/discussed.	3.85	Very Satisfactory
The exchange of ideas in the open forum between the speakers and the audience was useful in terms of further clarifying the current issues/concerns and in identifying possible solutions to some problems.	3.85	Very Satisfactory
I learned new information which can improve my knowledge and understanding of issues and concerns on nuclear science and technology	3.94	Very Satisfactory
The organizers were efficient, helpful, and accommodating.	3.90	Very Satisfactory
The venue is not appropriate for activities such as this.	1.93	Fair
The time allotted for the presentations and open forums is not enough	2.50	Satisfactory
The organizers provided sufficient information kits/handouts.	3.74	Very Satisfactory
This event is well-organized.	3.77	Very Satisfactory
Overall Mean Score for PNYS 2018	**3.60**	**Very Satisfactory**

In general, participants gave high ratings for the PNYS 2018 [1]. Participants stated that all the activities were relevant, that the speakers were able to clearly and effectively communicate their messages about different aspects of nuclear science and technology, and that they gained new knowledge by joining the PNYS 2018. On the other hand, some participants suggested to give more time for the event so that all participants could experience all the parallel activities. Other suggestions included PNRI facility tours and more interactive exercises and activities, as well as for open forums and talks and the need to organize the venue set-up to make it more conducive for the activities. Furthermore, participants also suggested that the activities be more environmentally friendly or produce less waste.

Aside from the evaluations, one activity of the PNYS 2018 which is the Nuclear Amazing Race was also shared to secondary school teachers from 14 countries in the Asia-Pacific region during a Training Course organized by the International

Atomic Energy Agency (IAEA) for its TC 0079 project last June 17-28, 2019 held in Manila, Philippines.

B. Evaluation of the Nuclear Youth Forum.

A total of 46 completed evaluation forms were collected form the participants, who were members of the PYGN Core Team. The breakdown of the PYGN Core Team present during the Nuclear Youth Forum is presented in Figure 3.

Half of the participants were college or graduate students who are also student leaders in their chosen organizations, who were chosen to aid the PYGN in establishing future collaborations. Almost 30% of the participants were teachers from senior high schools or colleges, which was also deemed advantageous as these members could serve as connections for possible talks or events on nuclear science and technology in their schools. The other young professionals were also chosen as participants for them to share their learnings on the beneficial applications of nuclear science and technology to their communities (officemates, friends, etc.).

Following the same format for grading as the PNYS 2018 evaluation forms, the scores from the 46 completed evaluation forms were averaged and are shown in Table III.

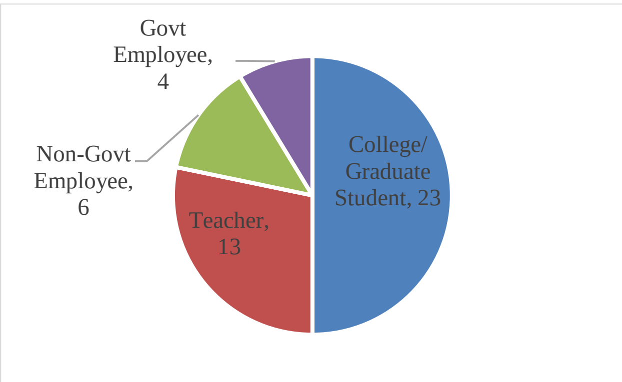

Figure 3. Number of Nuclear Youth Forum Participants per Classification

TABLE III. MEAN EVALUATION OF NUCLEAR YOUTH FORUM

Item on Nuclear Youth Forum Evaluation Sheet	*Mean Score*	*Descriptive Rating*
Relevance of Nuclear 101 Talks: Debunking Myths on Radiation and Radioactivity	4.00	Very Satisfactory
Relevance of Nuclear 101 Talks: PNRI R&D Projects on Nuclear S&T	3.96	Very Satisfactory
Relevance of Nuclear 101 Talks |Inspirational Talk for the Youth	3.98	Very Satisfactory
The speakers were able to clearly and effectively communicate the message to the audience.	3.94	Very Satisfactory
The topics were adequately explained/discussed.	3.93	Very Satisfactory
The exchange of ideas in the open forum between the speakers and the audience was useful in terms of further clarifying the current issues/concerns and in identifying possible solutions to some problems.	3.78	Very Satisfactory
I learned new information which can improve my knowledge and understanding of issues and concerns on nuclear science and technology.	3.91	Very Satisfactory
The organizers were efficient, helpful, and accommodating.	4.00	Very Satisfactory
The venue is appropriate for activities such as this.	3.96	Very Satisfactory
The time allotted for the presentations and open forums is sufficient.	3.62	Very Satisfactory
This event is well-organized.	3.78	Very Satisfactory
Overall Mean Score for Nuclear Youth Forum	**3.90**	**Very Satisfactory**

The team worked to improve time management and the set-up of the venue based on the suggestions given during the PNYS 2018. Improvement was also reflected in the increase in the average scores for the venue and the time allotted for activities. In general, the Nuclear Youth Forum received higher average grades in almost all the items of the evaluation form compared to the PNYS 2018.

C. Evaluation of the PNYS Video Making Contest

The 135 individuals/teams who submitted entries for the PNYS Video Making Contest were not asked to evaluate the contest as the organizing committee deemed that the results of these evaluations may be biased due to the results of the contest. Thus, the organizing committee evaluated the contest based on the number of likes, shares, and reach of the official entry posts on the PNRI Facebook page, and the implementation of the contest.

The minimum, maximum and average values for the likes/reactions, shares, and reach of the official entries are shown in Table IV.

TABLE IV. DATA OF PNYS VIDEO MAKING CONTEST FACEBOOK POSTS

	Minimum	*Average*	*Maximum*
Likes/Reactions	2	428.88	6200
Shares	0	1315.30	59000
Reach	474	6571.31	67900

Reach is defined by Facebook as the "number of unique people who saw the post". All the reported values are based on the organic reach of the posts, as none of the posts were boosted during the time of the contest.

Based on the Facebook likes/reactions, shares, and reactions, it can be concluded that the PNYS 2018 video making contest was successful in promoting the beneficial applications of nuclear science and technology as an average of 6,571 unique Facebook users were able to see the content of each video. A total of 290 students participated in the contest. The organizing committee also noted these 290 students learned more about the beneficial applications of nuclear science and technology to the young generation as thorough research on the topic is needed in order to produce the videos they submitted.

For the implementation of the contest, the organizing committee noted that it was difficult to manage the large amount of entries. This caused changes in the timeline stated in the mechanics as it was not possible to upload all 135 entries at the same time. Thus, the online voting of official entries was done in batches. All changes in the schedule were communicated to the participants via email and were posted on the PNRI Facebook page.

D. Evaluation of the Jingle & MTV-Making Contest

The 29 individuals/teams who participated in the Jingle & MTV-contest were also not asked to evaluate the contest for the same reasons as the PNYS Video Making Contest.

The minimum, maximum and mean values for the likes/reactions, shares, and reach of the official entries are shown in Table V.

Based on the Facebook likes/reactions, shares, and reactions, it can be concluded that the Jingle & MTV-making contest was successful in promoting the beneficial applications of nuclear science and technology as an average of 3,721.86 unique Facebook users were able to see the content of each video. A total of 158 students participated in the contest. It was also noted these 158 students learned more about the beneficial applications of nuclear science and technology to the young generation as thorough research on the topic is needed in order to produce the videos they submitted.

TABLE V. DATA OF JINGLE & MTV-MAKING CONTEST FACEBOOK POSTS

	Minimum	***Mean***	***Maximum***
Likes/Reactions	76	711.93	2700
Shares	23	10928.71	88082
Reach	1514	3721.86	7958

For the implementation of the contest, it was noted there were significantly less participants in this contest compared to the last one. The organizing committee concluded that this is because the task of writing an original song and making its corresponding music video is significantly harder than making a one- to two-minute video.

While autolikes or bots were not a problem in this contest, one post was blocked several times during the online voting. Despite warning all the participants, it is still beyond the control of the organizing team and this problem may still arise if Facebook likes/reactions and shares will still be part of the criteria for future contests. Despite these problems, the organizing team still concluded that Facebook likes/reactions and shares should still be a part of future contests as these allow more Facebook users to learn more about the beneficial applications of nuclear science and technology.

IV. CONCLUSIONS

Despite starting only in November 2017, the Philippine Young Generation in Nuclear (PYGN) in the partnership with the Philippine Nuclear Research Institute (PNRI), successfully implemented activities and contests that were able to promote the beneficial and peaceful uses of nuclear science and technology to young Filipinos.

Based on the evaluations of the Philippine Nuclear Youth Summit and Nuclear Youth Forum, the organizing teams were able to plan out relevant and effective activities where participants were able to learn more about nuclear science and technology.

Based on the Facebook likes/reactions, shares and reach of the official posts of the video entries, the PYGN and PNRI successfully implemented contests that were able to spread the beneficial and peaceful applications of nuclear science and technology to the student participants and to their Facebook contacts.

ACKNOWLEDGMENT

The authors would like to thank the Department of Science and Technology - Philippine Council for Industry, Energy and Emerging Technology Research and Development (PCIEERD) for funding the above-mentioned events and contests.

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Using Lessons Learned from the EU Referendum to better promote the Nuclear Industry

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I. INTRODUCTION

The United Kingdom has recently found itself paralysed by a number of political issues. The most prominent of these has surrounded the UK's membership of the European Union, but climate change continues to lurk as the most serious issue of the day [1]. The British Government has now committed to an ambitious emissions target. To achieve this requires collaboration and innovation from all sectors. It is believed that this can be achieved through a number of methods, with a minimal (1% of GDP) increase in real time spending when carried out effectively [2]. The research supports a fluid approach to solving climate change but recognizes nuclear as a large portion of that mix [2]. However, nuclear's role is uncertain, and would require large investment [2]. New build in the UK is the current option for replacing the requirements from offsetting coal and gas by 2030, with the ambition that small/advanced reactors can begin to be implemented in the interim (in the UK a small reactor is a near term modular reactor, whereas an advanced reactor constitutes anything falling under the Generation IV tag) [3]. Developments in modular build and fuel efficiency could provide nuclear with a huge advantage over renewables, whilst the potential for Nth of a kind scenarios with small reactors could dramatically reduce the main cost- the construction [3], resulting in nuclear having the capability to play a major role in the UK energy mix. However, this can only take place with support from Government [2].

II. PUBLIC ATTITUDES TOWARDS NUCLEAR ENERGY

Government officials do not plan long term and will always prioritize quick wins due to the cycle of election and cabinet reshuffle causing their positions to be impermanent [5]. The best a UK minister can hope for under a stable Government is five to ten years of policymaking on a given subject. Compared with the lead times for development and the political turmoil, it is hard to imagine policy makers signing up to complex programs of research and innovation supported by their successors and ultimately running risk of their foresight being credited to another party. Therefore, public support is required for long-term investment.

Issue	%
Energy prices	33
Renewable energy sources	14
Electricity supply	12
Limited energy sources	9
Nuclear energy	8
Environmental issues	7
Energy consumption	6
Fuel	6
Ways to use energy	4
Gas	4
Energy dependence	3
Importance of energy	3
Other fossil fuels	3
Power plants	1

Figure 1- Poll of the biggest issues felt by the UK public regarding energy supply. Showing percentage of people who identified each issue as the largest concern. [4]

Public support for nuclear remains steady. Public attitudes [6] found 40% in support of nuclear, 22% against and 37% being neutral. Generally, there has been agreement that nuclear could provide safe, affordable energy that helps reduce carbon emissions [6]. Positive responses are highest amongst males, the highly educated and those leaning towards the right of the political spectrum [4]. More people are concerned about the price of their energy and the security of supply than they are what source it came from, [4] and a lack of understanding of how waste is managed [6]. Finally, it is useful to understand that nuclear 'Incidents' such as those at Fukushima tend to have big immediate impacts that gradually reduce [7]. The threat from radioactivity can form a perfect storm for the uninitiated public; they cannot see, smell, taste or hear it, so it cannot be anticipated or avoided. Add to this the long latency effects of nuclear exposure and the unavoidable parallel between energy and weaponry, layered with the perception that any understanding of nuclear physics is beyond them and you have a subject that people feel drawn to have a negative perception of, without any counterfactual.

III. WHY IS THERE NOT MORE SUPPORT?

Still, despite the benefits of nuclear power the fact that support is low is a concern when it comes to getting government buy in. To understand why a good example to look at would be the reaction to the recent HBO series 'Chernobyl'. After the series there was a huge upswing in internet searches looking for information suggesting a desire for more knowledge on the subject and a better understanding of how it has affected modern nuclear plants. However, the creator commented that his intention was to show how a lack of adherence to safety culture and subsequent cover up was the real reason for the disaster [8]. Plants contain an inherent risk (as does every other industrial process) but the safety culture and regulation associated with plants (especially in the UK) is second to none. There is a lack of knowledge of nuclear processes amongst the UK population. The UK school curriculum core is power generation but no time is given to radioactive waste storage/disposal, medical isotopes, or how radiation works. This makes it easy for anti-nuclear groups to take advantage of lack of knowledge and sentiment to whip up fears amongst the public. Combine this with the tabloid press' desire for a headline and stories such as 'Worker Exposed to Sellafield Plutonium had Skin Removed' [9] and 'NUCLEAR FEAR- Security Scares at Sellafield nuclear waste plant raise fears of disaster 'worse than Chernobyl' [10] only serve to stir up hysteria amongst those who do not have a grounding in the subject. The nuclear industry, unfortunately, does not offset this with positive press. The UK's Nuclear Decommissioning Authority is always ready to promote how risk from another dangerous part of a site has been reduced through its clean-up efforts, but when in front of a Parliamentary Committee in 2018 to defend its £3.2 billion public budget and £900 million worth of cancelled projects [11] the Nuclear Decommissioning Authority failed to articulate that the cancellations had saved billions in the long run. People are more worried about the effect on their everyday lives than they are the long-term safety hazards of nuclear energy and the industry needs to 'get a grip' on this if it wants to promote a positive light.

All of this reflects that more has to be done to better promote the benefits of nuclear- The UK Energy Research Centre noted that despite no fundamental changes, the re-framing of it as a tool in the fight against climate change has seen an increase in support, with a large decrease in the number who feel the risks outweigh the reward [7]. This minor change in approach would address the real issues people are worried about as shown in [6] and tie nuclear into a more positive message for the future. Studies note that ambivalence towards the industry makes up for a large percentage and being able to sway these towards the 'in favour' side of the argument could be crucial to sustained support. The more understanding and exposure people have of nuclear power, the more likely they are to support its use - as evidenced by countries such as France which has high capacity for nuclear and sees strong support from its citizens [4]. This is backed up by the knowledge that the biggest concern currently with nuclear power is in the storage of waste- due to a lack of understanding of current and future practices [7]. The less people know about something, the more likely they are to distrust it which is having a huge impact on the benefits nuclear could provide.

This is not a problem that is localised to the British. One article [12] reflected that only 21% of Americans felt well informed on nuclear power and that 70% did not know that it produced more energy than wind or solar. Following this, when told that it is responsible for 63% of US low carbon energy and was the only low carbon source that could be kept on indefinitely without emmiting greenhouse gases 86% of those surveyed agreed that it should be more prominent (of these, 88% reflected a direct change from their original opinion). This familiarity breeds positivity, with positive views of nuclear being 16% more amongst those that live in close vicinity to a power station. The message here is that getting people to engage with the realitites provides the positive side of the argument offering people a chance to make their own decisions.

IV. PARALLELS THAT CAN BE DRAWN FROM THE UK REFERENDUM AND THE NUCLEAR INDUSTRY

The recent referendum on the UK's membership of the European Union presents parallels with how we should promote the nuclear industry. The subjects involved are similar in that they were both background issues until they were brought to public consciousness, and both featured a large group who were neutral- the winning campaign won by a margin of less than 4% [13]. Areas of the country with low investment and education levels were more susceptible to vote leave because they had more undecided voters [13]. The reason leave won over remain, was down to two reasons. One, they accepted that there were groups that would vote regardless, therefore it was best to focus on the undecided center ground and two, the campaign provided a positive message about leaving, whereas remain focused on the dangers and did not really provide a positive reason to stay [15]. The remain campaign ran on a platform now referred to as 'project fear'. Years of austerity following the financial crisis had resulted in a diminished trust in officials. The government's decision to not provide clarity in its decision making resulted in a number of votes that went against it- this draws parallels to a site such as Sellafield, where workers have expressed distrust at licensees over their refusal to share information [16]. Many UK papers also backed the leave campaign. Eye catching headlines were an ideal way to attract readers and paved the way for sensationalist articles that contained heavy anti EU bias and stoked fears- such as those detailing threats to the UK foods industry [17]. The British Press tends to lean heavily one way or another- therefore there is a requirement for nuclear to be positively viewed if it is to get the press coverage it needs. A recent documentary [18] theorised that middle ground voters in the UK were targeted through secondary messaging. Messages such as the warnings about immigrants fraudulently seeking unemployment benefits are perfect examples. The secondary message of benefits claims increasing taxes hit home in a country that was worried about paying its bills. Similarly, whilst £350m given to the EU didn't matter to the average consumer, £350m to the NHS would have provided a boost to service and quality of care.

V. Applying the Lessons from the EU Referendum to the Promotion of the Nuclear Industry

Knowing what we do about how the referendum was won we can start to apply this learning to promote the nuclear industry and provide some steps to better endorse nuclear in the UK:

- Increase knowledge levels. The government needs to ensure that students understand where energy comes from, this includes providing fair coverage to all energy generation methods and also promoting science and technology to students of all backgrounds. Older students need to be encouraged to better understand radiation uses and hazards to form their own opinions. Industry representatives currently do a great deal with their local communities - but more should be done to educate on a wider scale. Additionally, sites need to be open and transparent ensuring trust in their capabilities does not allow dissent to diffuse through the local communities- as seen in Cumbria (16).

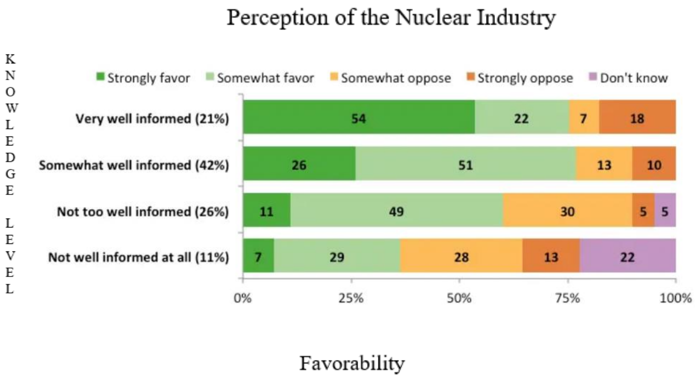

Figure 2- Favorability towards the nuclear industry based on perceived knowledge level. [12]

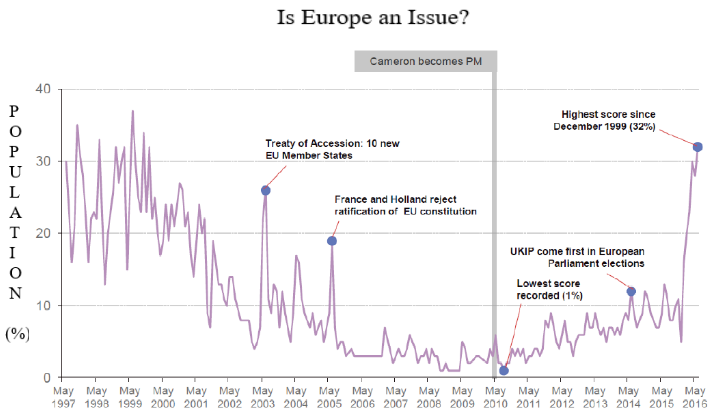

Figure 3- Graph showing percentage of people that identified Europe as the top issue facing the UK. At one point in time less than 5% of people thought Europe was an issue before the referendum happened despite the result [14]

- Positive Press Coverage. The industry should cultivate good relationships with the press to provide the types of headlines they are looking for - Headlines such as 'Four times as many jobs created in Nuclear than Coal last year' or ' Nuclear provides almost DOUBLE the amount of energy as Wind' (19) will catch the eye, and shift public opinion.
- Focus on real issues. Information about nuclear needs to focus on the public's concerns – energy supply and jobs. The Rolls Royce Small Modular Reactor is planned to be a 440-Megawatt power station. This means nothing to the average person, but the same station being able to power the entirety of Leeds or 40 million lightbulbs [20] is a more enticing prospect to the public.
- Attack is the best form of defence. Many anti-nuclear groups oppose this and cite the dangers of nuclear rather than the benefits of wind and solar. The industry needs to fight fire with fire. The industry should start looking at why other technology cannot replace nuclear - lines like 'onshore wind requires an area of land roughly equal to the entirety of green spaces in the United Kingdom' [21] would do a much better job of dispelling the myth that nuclear can be ignored. It is no good comparing nuclear to renewables, it must carve out in own place in the energy mix.
- Focus on neutrals. Nuclear is likely to be supported in places such as Copeland [16]. Areas that are more likely to be swung by positive reinforcement

messages are neutral where there is minimal nuclear employment resulting in a lower dependence or likelihood of fightback.

VI. Conclusion

It is vastly important that we start promoting the nuclear industry smarter, better, and with more thought to targeting to ensure it maintains its place on the public agenda. A lack of investment in nuclear will lead to other local solutions overtaking the industry [2]. Using the lessons of the referendum can provide us with ideas for a pathway to deliver this and we now have a better understanding of the public mood which can be taken advantage of. Without public support nuclear will not attract investment from a government that wants to keep its voters on side for future elections, and so it is up to us to ensure it remains on the agenda.

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Human Resource Development Cosiderations and Challenges for Nuclear Power Programme of Bangladesh

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I. INTRODUCTION

The development of an effectively competent manpower is one of the fundamental conditions of success for any Nuclear Power Programme. Without capable manpower no nuclear power plant can be planned, built or operated appropriately, and nuclear safety and reliability of power production cannot be assured. The amounts and qualification of manpower required for a successful nuclear power programme are usually underestimated and the resulting shortage of manpower is a restraining the development of nuclear technology in developing countries. Before embarking on its first nuclear power project, it is essential for a newcomer country to determine its real manpower needs in the framework of the envisaged nuclear power programme and to evaluate the existing organizational, educational and industrial capabilities for meeting these needs. Manpower development for domestic participation in the nuclear power programme should be considered within the broad context of the national industrial development strategy and its overall manpower requirements.

The recent TMI Nuclear Accident has reaffirmed the importance of qualified manpower for safe operation of nuclear power plants. Although a combination of several factors contributed to this accident, human errors played an important role in its development and in its ill-control. After TMI, the nuclear power plants have expanded their operator training programs. Plants have also modified their control room indicators, and have modified some plant equipment to prevent other accidents from occurring.

Adequate measures required to prevent an accident can only be ensured by operator skill and proficiency, achieved through appropriate education, training, experience and systematic measures for maintenance of proficiency. The essential role of skilled and qualified people in preventing accidents or handling them properly if they do occur has been once again confirmed.

Nuclear power, a demanding technology, requires specific knowledge and excellence in human performance. Education and training takes time and effort, and a nuclear man-power development programme must be planned a long time in advance to be effective. There is a fundamental need therefore for a country embarking on a nuclear power programme to provide for the establishment and implementation of a manpower development programme that covers all aspects of manpower development including determination of needs, both in respect of manpower qualification and numbers, a programme of basic and specialized education, programmes of specialized training and qualification through experience.

II. NUCLEAR POWER PROGRAMME OF BANGLADESH

Bangladesh began its preparations for nuclear power in the 1960s and Rooppur Nuclear Power Project (RNPP) site was selected in 1963. But since then, the progress never passed the pre-implementation stage. Bangladesh government has a firm determination to implement nuclear power plant in near future to help meeting the country's increasing energy demand. A resolution for immediate establishment of NPP to combat existing power crisis was passed by the Parliament in 2010. Power system Master plan 2010 of Bangladesh Government indicates that Bangladesh government wants to introduce 2000 MW of Nuclear power by 2020. Addition plans of another 1000 MW in 2025 and further 1000 MW by 2030 has also been included in the power system master plan. On 2nd November 2011, Bangladesh signed an Inter-Governmental Agreement (IGA) with Russia for a Nuclear Power Plant (NPP) with two units, each of 1,200 MWe. Under the IGA, Russia will also provide financial support, support for human resource development (HRD) for the first NPP in Bangladesh, supply the nuclear fuel and take back the spent nuclear fuel. The target for commissioning the first NPP and starting commercial first unit will be commissioned in 2023 and the second in 2024. Bangladesh Government has also decided that NPP will be implemented through Engineering Procurement Construction (EPC) turnkey contractual approach. Bangladesh recognized that the nuclear power programme is a major undertaking requiring careful planning, preparation and

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investment in time and strong government commitment of at least 100 years; the development of such a programme entails attention to many complex and interrelated issues over a long duration. The introduction of nuclear power requires the establishment of a sustainable nuclear infrastructure that provides governmental, legal, regulatory, safety, technological, managerial, industrial and institutional support. Experience also indicates that the competent HRs usually constitutes the most crucial constraints in the development of nuclear power and implementation of a NPP project. The manpower development for a nuclear sector is a long-term activity. Bangladesh has made a clear commitment and taken a knowledgeable decision for creation of nuclear power infrastructure to build Rooppur NPP. As a newcomer country Bangladesh has put much effort in creating national conditions for construction and erection phase activities according to the IAEA Milestone Approach (IAEA Guidelines NG-G-3.1) as shown in "Fig. 1" [1].

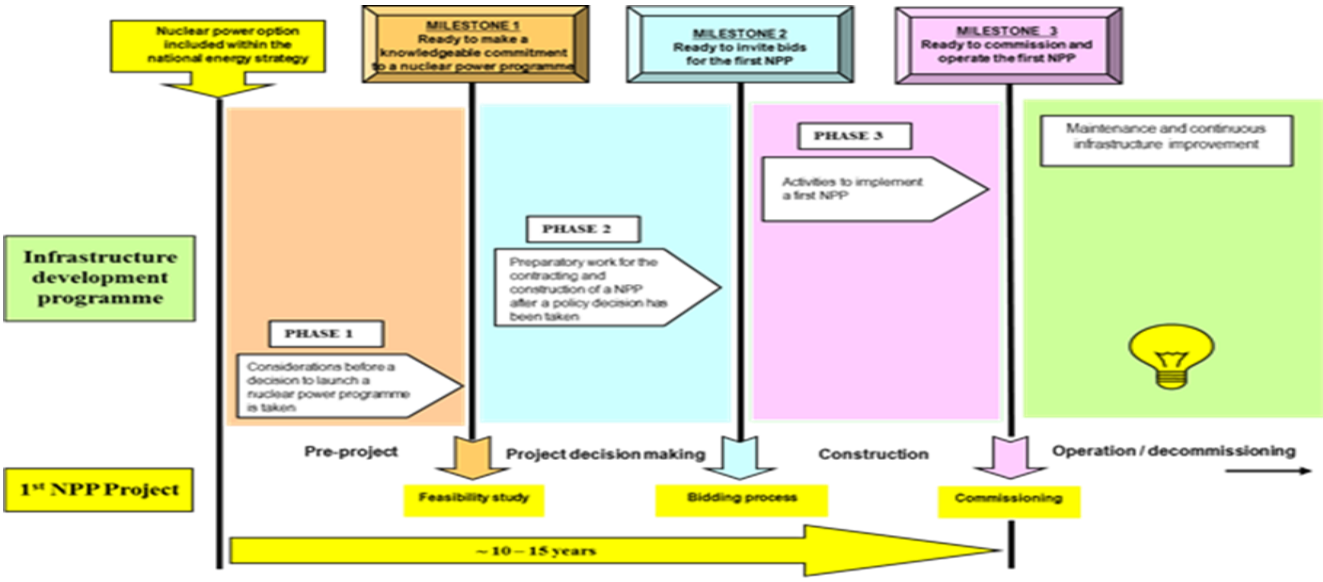

Figure 1: Infrastructure developmentprogramme according to IAEA Milestones Approach (Nuclear Energy Series, No. NG-G-3.1, IAEA, 2007)

III. HUMAN RESOURCE DEVELOPMENT

A. Overview

Development of human resources is a critical infrastructural issue to the countries embarking on their first Nuclear power plant. Bangladesh has lack of the required infrastructure, technical and professionals for construction, operation and maintenance of the Rooppur NPP. The main reason of this lack of competence is the lack of adequate education and training in The field of nuclear science and technology in the current national education system. So it is clear to build up a capable manpower for the first NPP we have to rely heavily on the vendor country for necessary education and training. But It is really important that besides developing an effectively competent workforce for the first NPP, we should also focus on developing a long term strategy of human resource development so that Bangladesh can enhance its current education and training systemso that Bangladesh can be self dependable in producing the next generation of workforce for Rooppur NPP and also the workforce for successful implementation and operation of the next Nuclear Power plants. A well planned Human resource development Strategy can play a vital role in a sustainable nuclear power programme for Bangladesh. Considering that a turnkey approach for Rooppur NPP construction will be used, the country is developing an approach to the shaping of human resources for the entire nuclear programme aiming to prepare for, build and operate the country's NPP safely and successfully.

B. HRD Needs: Workforce Planning

The workforce planning process for nuclear energy programme must begin with a view of the activities identified under each of the 19 infrastructure issues, phases by phase, which should form the basis for an initial analysis of competence and resources needs. Workforce planning and training of the staffs of these two key organizations are critical issues. Moreover, a strategic HRs plan for capacity building in the national nuclear education and training infrastructure has to be established to prepare for and build a NPP and also manage and operate the plant in safe and sustainable manner [2]. Whilst significant planning of human resource development is needed in the earlier phases, the main task of staff development for the operating organization takes place in PHASE 3. At the start of the phase, they are mainly a project management organization and by the end of PHASE 3 all the necessary human resources should be in place to commission and operate the first nuclear power plant. In addition, educational and training programmes to develop a continuing flow of qualified people to all areas of the programme should be well under way, and the government should continue to promote educational and industrial development for national participation in the nuclear programme. Personnel of the key organizations during three successive phases of nuclear power programme is shown in “Fig. 2”.

D.Recruitment

The mass joining ceremony took place in an elaborate function at the project site of Rooppur in the Pabna district. With the joining ceremony, the country’s first nuclear power plant got its first batch of Bangladeshi scientist and engineers. The newly joined professionals will eventually get training from Russia and India.

C. Selection, Education and Training of Rooppur NPP

Bangladesh Atomic Energy Commission (BAEC) has a good number of professionals in various branches of nuclear technology; The Core Manpower for preparatory activities of the Rooppur Nuclear Power project is available in BAEC. Fresh professionals for RNPP and the nuclear regulatory body will be available from the General Universities, Universities of Engineering and Technology and Technical Institutions from their annual output.

A general approach of recruiting manpower for RNPP year by year during different milestones has been proposed on the basis of the following issues:

- Different work functions
- Qualification of manpower and
- Experience and Training requirements for different work function categories
- The actual work functions, their basic education criteria, lead time, and no of personnel required is dependent on technology provided by vendor and socio-economical condition of owner country.
- An extensive Feasibility Study on HRD will be carried out for Rooppur Specific Information to Russia and India for training, according to the contract for development of human resources. [Ref 4]

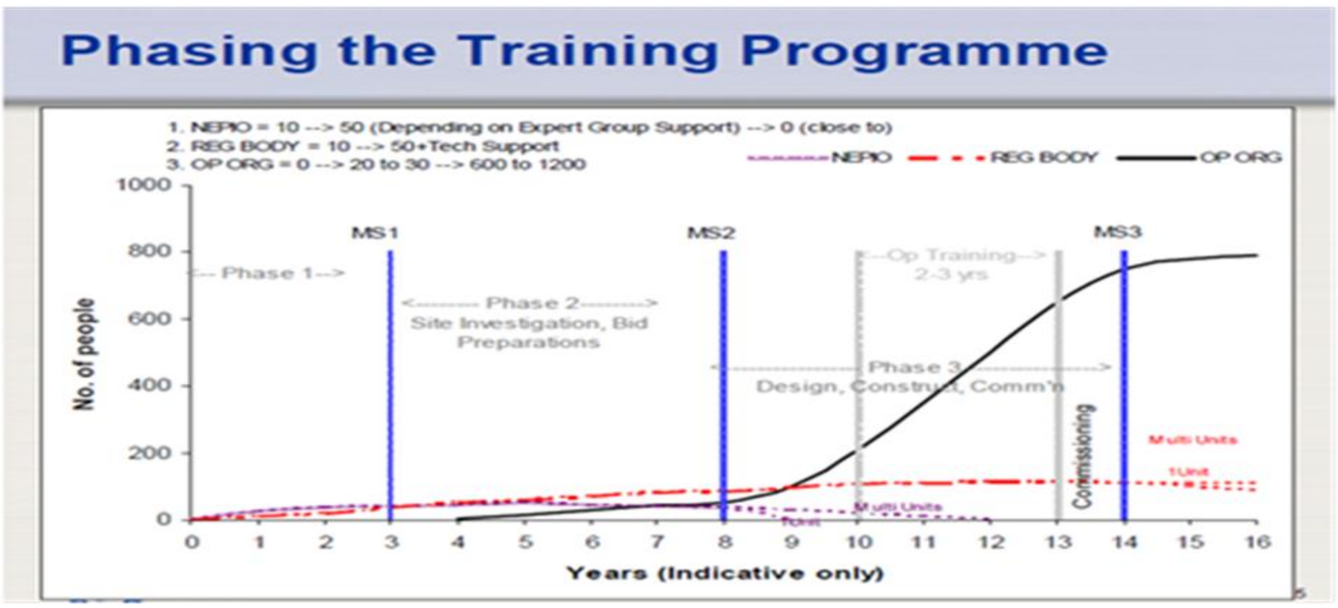

Figure 2. Personnel of the key organizations during three successive phases of nuclear power programme.

After that they will take charge of the RNPP. A Joint Training Advisory Commission (JTAC) is formed by Bangladesh and Russia to perform operations like manpower selection, training, recruitment and quality examination [4].

- Recruitment will be done in phase according to IAEA Guidelines
- Personnel requiring more education and training will be recruited earlier
- BAEC has identified that an approximately 1660 personnel will be required for NPP operation and maintenance. This manpower will be recruited in different phases during the period 2011-2023. An initial prediction of the number of people to be recruited in different phases has been predicted based on international approaches and practices. It is observed that our estimation follows the traditional IAEA S-curve approach [3].

Engineers are divided into their respective branches. Twenty professionals are being appointed in civil engineering section along with 28 in the mechanical department, 20 in electrical section and two in the chemical engineering department. Among the scientific officers, 13 people have been appointed in the department of physics along with four in applied physics, five in nuclear engineering and two in chemistry. In addition, two persons have been appointed in the Human Resources Department along with two in the Department of Finance and Accounts and two in the Environment Department [4].

E. Education and training

- Education and training of man power for NPP projects (siting, consulting work, project management, operation and maintenance etc.)
- Education and training of man power for long term nuclear program (regulatory body, staff for universities, scientists for Research and Development)
- The skill of these scientists and engineers will be developed to the international standards for the construction and operation of this power plant safely and environmentally under the general agreement of the nuclear program. These scientists and engineers have been sent already in Russia, India and will be sent

F. Proposed HRD through vendor and international co-operations

- RNPP-Oriented Human Resources
  - Project promoting and implementing manpower (professionals, technicians for technical description, design aspects & approval, construction)
  - Operation & Maintenance (O&M) Manpower (professionals, technicians and craftsmen)
- NPP-Program Oriented Human Resource
  - Education, Research and development in nuclear S&T professional
  - Nuclear Law and regulation professionals.

IV. RECOMMENDATIONS

From the above discussion it is clear it's the appropriate time to Bangladesh to consider some serious action to plan an integrated Human resource development plan. In order to develop an integrated HRD plan the following actions should be considered.

- A feasibility study on HRD should be carried out to make to find out all details of the current national educational system. The study will provide us the information of what type of education is available, the syllabus and course curriculum, number of students passing each year and their competency level.
- From the results of the feasibility study a comparison can be made with the vendor countries requirements for nuclear power plant staff. From this comparison we can identify the gaps of our education system.
- Once the gaps have been identified in collaboration with the vendor a specific action plan should be developed to fill up the gaps.
- For the first NPP a detailed training program should be developed where the fresh graduates and technicians will get further education and training necessary to meet the requirements for the NPP.
- In collaboration of universities of vendor countries local universities should consider upgrading their syllabus and course curriculum so that gradually most of the education can be completed in local universities thus decreasing the dependency on education in foreign universities. Also the necessary instructors for the new courses should be developed.
- A local Nuclear Training Institute with adequate facilities and trainers should be setup so that in future
- Reliance on foreign training can be lessened.
- Development of the different Technical Supporting organization should be considered as a part of the whole Nuclear Power Programme.

V. CONCLUSION

Bangladesh faces a huge challenge in implementing the Rooppur Nuclear Power plant and its total nuclear power program. The IAEA will continue to support the country in the areas of safety, regulatory framework, management, human resource development, security, safeguards and emergency planning. A well planned and executed human resource development strategy can make it easier for Bangladesh to implement the Rooppur nuclear power plant successfully and in the long run make sure that we can continue a sustainable nuclear power program. Here a preliminary assessment has been chalk out a plan to find out methods that can be applied to find out our gaps and then fill them up. Further work is in progress to prepare a realistic and detailed human resource development program for a sustainable nuclear power program of Bangladesh.

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The *Dos and Don'ts* of Science Communication

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I. Introduction

Communicating science to the public can be difficult. Communicating nuclear science to the public is even more challenging. Even so, it is crucial to discuss the peaceful uses of nuclear and radiological technology and to engage in conversations. Avoiding technical details and jargon and combating the many myths and misconceptions that exist in these areas amongst the general public is essential. We are ANSTO employees and are part of the 2019-2020 cohort of the "Superstars of STEM" program (Figure 1). Superstars of STEM [1] is a program run by Science and Technology Australia that aims to smash the stereotype of who an engineer or scientist is through supporting women in STEM to become visible public role models. In order to do this, the program equips female scientists, engineers and technologists with advanced communication skills and provides them with genuine public opportunities to use these skills. These skills are vital when we tackle nuclear science topics when communicating with the public. Through a series of hands-on workshops, Superstars of STEM covers many aspects of communication from storytelling to media training to communicating with influence. This paper is inspired by TEDx producers, who provided an insightful session on how to give a public TEDx talk on science as part of the Superstars of STEM personal brand and storytelling workshop. We will cover many tips and tricks shared at the workshop; however, some points are also common sense – we've all been to many, many presentations as an audience member and, in general, know what makes a good or bad presentation. Here we will focus on the dos and don'ts of presenting to, or communicating with, a public audience. We hope to provide practical tips on how nuclear professionals can maximise the impact of their public communications. It highlights how the audience plays a primary role in dictating the style and approach for nuclear communication.

II. The *Don'ts* of Science Communication

A. *Death by PowerPoint*

A number of years ago, one of the authors was requested to review a presentation that was targeted at a group of nuclear professionals. The presentation was "only" 50 slides and was expected to take a few hours to deliver. The presentation was opened only to find that not one slide contained an image. Fifty slides without a single picture! It was 50 slides of text-intensive bullet points; page after page of text. Every single slide looked the same. It met the definition of 'death by PowerPoint'. The unfortunate people on the receiving end of this presentation would need at least five coffees to stay awake.

When subjected to a presentation of this sort (and let's face it we have all sat through many of these kind of presentations), what are you supposed to do? Read the many dot points and try to make sense of them? Listen to the presenter read the dot points out word for word? People are likely to scan a few words on the slides for context before trying to give their attention to the presenter. And if the presenter isn't engaging? Well, we've all been there right? Good thing there are these things called smart phones. Or, if you're too polite to start playing with your phone, there's always the anticipation that the next slide will not be exactly the same as the last 35….

With a bit of thought and creativity, any presentation can be transformed from a 'death by PowerPoint' experience to one that engages. We will go through some dos of don'ts of presenting but the number one thing NOT to do during ANY presentation is to bore people with bullet points or lengthy chunks of text.

Figure 1. Superstars of STEM 2019-2020 cohort (Joanne Lackenby on the left, Katie Sizeland on the right).

B. Jargon

You are a nuclear expert. You know what 'activity', 'barn', 'fertile' and 'breeder' mean in the context of your profession but imagine you're a member of the general public. You hear a presentation about the activity occurring in the barn involving breeding. What the audience is now picturing will almost certainly be a world away from the topic of your presentation.

Cut the jargon. If your grandparents or neighbours don't know what a word means, get rid of it. Find a better way to explain it and be creative. You may be the type who likes to be technically correct all the time. The public won't know if your analogy or replacement words without the jargon aren't perfect. If the audience doesn't understand what you are saying, you won't achieve the objectives of your talk (whatever they may be). Save the jargon for your colleagues. Impress them with your knowledge of fertility, not a public audience.

C. Slides

Slides can be very beneficial for a public audience, especially for technical topics. However, there are some definite 'don'ts' when using slides in public.

Don't use a slide unless it is absolutely adding value to what you're saying. Remember, the audience is there to hear you speak and learn from you. They are not there to read your slides. So if a slide doesn't add anything to what you're saying, delete it.

Don't use slides as reminders of what you want to cover in your talk. You can definitely and easily avoid having to do this. If you need something to prompt you, use cue cards instead of slides to remind you what is coming next. This means you need to be prepared. The easy answer? Practice, practice, practice. Slides should be for the benefit of the audience, not the benefit of the presenter.

We have all seen many presentations that overload the audience with an abundance of animations and PowerPoint effects, where every word spirals into view with a different sound effect each time. Even without endless animations, an audience can still be hit in the face with overly numerous fonts, word sizes and colours. The morale of the story here is to make sure you keep the slides as simple and clean as possible. If the slides are too 'busy', all they will do is distract the audience and subtract from your message.

D. Timing

Time management is important for any presentation. If you have been given 20 minutes, then stick to it. Show respect for the audience and any speakers who follow you. Flipping through slides because you've run out of time is another big 'don't'. It shows you are not organised and that you haven't given the presentation the preparation time it deserves. Focus on having fewer slides and making them of professional quality. In this way, you can deliver high quality information rather than trying to rush through one slide every 10 seconds, because you haven't thought about timing. This comes back to the same easy answer as above. Practice, practice, practice.

Another top tip if you have any influence on the order of proceedings: if you think your presentation will generate a significant number of questions, try to go last. That way people who have heard enough can leave, and those with burning questions can still ask them.

III. The *Dos* of Science Communication

A. It's all about your audience (not you)

Talking to the community about nuclear topics can be difficult. Content is often technical and the audience may have little knowledge in the area. You may even encounter common misconceptions or biases the public have about nuclear topics. Therefore it is vital that in your talk you consider the needs of the audience and not your own. You may want to convince the audience that nuclear power is the best thing since sliced bread. To do this, you need to understand your audience. You need to know who your audience will be and you need to understand what the audience is interested in or concerned about and focus on this. Spouting all the wonderful facts you know is unlikely to get you anywhere. If possible, research your audience beforehand to try and determine why they are in the room and what they want. That being said, you also need to know what you want to achieve. Whether you want to build your skills in public speaking, convince the audience of a position, start a discussion, or something else, define your objectives.

Lastly, empathy goes a long way, especially if you're trying to persuade the audience of the merits of your topic. Answer a question "what about the waste created by nuclear?" or "is it safe?" respectfully and patiently even though it may be the 1000th time you've answered the question.

B. So What?

In order for any presentation or communication to be engaging and effective, it absolutely must address the question "so what?". Why should the audience care about what you're saying? What's in it for them? You need this connection for your presentation to have impact. Again this comes back to knowing who your audience is. If you can get people to understand early on why they should be listening to you and why the content applies to them, you can have more emotional impact and the communication will be much more effective. Remember - it's not about you, it's about them.

C. Grab attention early and tell stories

"Hi I'm so and so and today I'm going to talk to you about nucleonics codes for nuclear reactor design". This statement may be factual, but it isn't inspiring. Try and think of a sentence or two that will really grab the audience's attention. For maximum impact, you need to bring people on a journey and you need them to start their journey from the very first sentence. An anecdote about your work or a story related to your message will be much more memorable than facts and technical details. Telling an audience about how you set up your neutron beam experiment from the technical perspective may inadvertently put some people to sleep. However, if something memorable happened while you were setting up your experiment that people may empathise with, tell that instead. You can still get the same message across, just in a different way. Work stories into your presentation wherever possible and appropriate. A story about how your research resulted in saving the life of a 2-year-old is far superior than giving out statistic after statistic about your research. The big picture is still needed, but people relate to stories not statistics or numbers.

D. Slides

Slides are a double-edged sword. They can be good (enlightenment) and evil (boredom and restlessness). So when is a slide a "Do"? Slides are very valuable when they contain images, but not just any image. Brought to light at the Superstars of STEM workshop by TEDx producers, images should reveal, explain or delight. Images can capture new findings, explain difficult concepts or make the audience feel good (or bad). But don't overuse pictures, just as you shouldn't overuse words. When creating your slides, try and limit the number of words per slide. Ideally, see if you can reduce the number of words on each slide to six or less.

E. Be Authentic

Last but not least, you need to be yourself when presenting. "Fake it 'til you make it" won't work. If you want to establish trust between yourself and the audience, you need to be your authentic self. If you want to inspire, you need to be human. Be honest and sincere. It may even mean admitting you don't know an answer. If you come across as fake no one will believe what you're saying, and you may do more harm than good. You may even inadvertently compound public misconceptions about nuclear.

IV. Conclusion

Communicating science to the public can be difficult and communicating nuclear science to the public is even more challenging. Even so, it is crucial we engage with the public and tackle these conversations. We have outlined some of the dos and don'ts of science communication shared with us throughout the Superstars of STEM program. Much of this paper was inspired by TEDx producers and the insightful session they provided on talking to the public about science. While we covered a few dos and don'ts of science communication, this is by no means an exhaustive list and many points come down to common sense, too. So charge ahead. Take up every opportunity you possibly can to communicate on nuclear topics with the general public. Don't have slides full of bullet points, don't use lots of jargon, don't mistime your presentation or overwhelm the audience with as many animations, fonts, colours and sounds as you possibly can. Do focus on your audience at all times, do address the question "why should I care?", grab attention early and tell stories, do have beautiful images to reveal, explain or delight the audience and always be authentic. With great science communication we can tackle misconceptions and educate the public about nuclear topics. So now we challenge each and every one of you to dare to take on public engagement and discussions. Harness all of the dos and don'ts in this paper to present in the most engaging and impactful way possible. Together we can make a change. Together we can have a real impact. So go on….challenge accepted?

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Assessing the Importance of Public Awareness and Education on Nuclear Applications in the Diagnosis and Treatment of Cancer in Lesotho.

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I. Introduction

As the former International Atomic Energy Agency (IAEA) Director General Yukiya Amano alluded in his speech while addressing the International Conference on Nuclear Science held in Bangkok, Thailand in August 2016, Nuclear applications are indeed quite beneficial in various spheres of people's lives. These include, medicine, food production and energy generation as well as in many other sectors.

However, a great number of people globally still panic at the mention of the word nuclear as they recall the wretched events of Three Miles Island and the Chernobyl that caused adverse damage on humanity and the environment.

"Changing entrenched public opinion against nuclear science and its applications can be an up-hill struggle and can be a hard, long and expensive and sometimes demoralizing task. However, if nuclear science and its applications has to have a long-time future, the community has to make every effort to change public opinion in its favor (H. Oberhummer and J. Deutsch, "Public Awareness of Nuclear Science Why and How").''

This paper seeks to find out the best possible ways of educating and raising public awareness on the peaceful uses (benefits) of nuclear science in the health sector.

II. Review

Public awareness and education are essential to changing social and cultural norms. Both formal and non-formal education are indispensable in changing people's attitudes so that they have the capacity to assess their concerns on issues surrounding them. Knowledge of the applications of nuclear science and technologies outside power production is very little. In health and medicine however, many people are aware of the use of radiation and radioisotopes for diagnosis and treatment of various medical conditions, most commonly in cancer treatment. According to the Canadian Nuclear Association' s 2016 publication on The Role of the Nuclear Industry in The World it is very important to foster public acceptance and awareness about the societal and economic benefits of nuclear science and technologies and to build trust on proper management of nuclear material and technologies in use.

Aykol. F et al ,2002 further asserts that nuclear power utilization sustainability into the future, globally, is highly dependent on public perception. In the health sector, nuclear science plays a very critical role in cancer prevention, early detection and treatment therefore it is vital that people get educated to know about all available options.

Incidences of cancers are on the rise and are one of the main causes of death worldwide. This disputes the impression people used to have that cancer exists only in developed countries. Lesotho is also experiencing a rise in cancer cases. In 2018, the International Agency for Research on Cancer of the World Health Organization (WHO) reported 1888 new cases, 1335 deaths and 3268 3-5-year prevalence cases of cancer in Lesotho. The report further shows that there are more cases in women than men. Therefore, in designing public awareness and education campaigns, special attention should be paid to address the most at risk population categories and also to the most prevalent cancer type. Table I shows the adult risk factors to cancer in Lesotho as reported in the cancer country profile of WHO in 2014.

TABLE I. ADULT CANCER RISK FACTORS IN LESOTHO,2014

Risk Factor	*Males*	*Females*	*Total*
Current tobacco smoking (2011)	. . .	. . .	. . .
Total alcohol per capita consumption, in litres of pure alcohol (2010)	10.8	2.5	6.5
Physical inactivity (2010)	5.8%	6.3%	6.1%
Obesity (2014)	3.3%	20.4%	11.9%
Household solid fuel use (2012)	. . .	…	62.0%

The above identified risk factors should be incorporated into health policies as well as in public awareness and education programme designs such to address the groups that are at high risk.

Through the use of nuclear medicine and radiotherapy, a lot has been achieved in the cancer field over the years and many lives have been saved. It is therefore, essential to enhance public awareness on risks and causes, symptoms, diagnosis and treatment options available in order to beat cancer. For instance, it is important that people know about the different kinds of cancer (breast, cervical, prostate and many more) and prevention measures. and to encourage people to regularly test for cancer in order to have early detections hence higher chances of healing.

In Lesotho, the Papanicolaou (Pap) smear exam is used to screen for cervical cancer, the leading cause of cancer deaths among women in the developing world. However, statistics reveal that less than half (47%) of women age 15-49 have heard of the Pap smear exam. Eleven percent have ever had a Pap smear, and 4% have had a Pap smear in the past 12 months (Lesotho Demographic and Health Survey, 2014). Another trend discovered in the survey is that; while the proportion of women who have heard of the Pap smear exam has increased from 31% in 2009 to 47% in 2014, the proportion of women who have had a Pap smear in the past 12 months has decreased slightly from 6% in 2009 to 4% in 2014.

H. R. Sahar et al,2017 suggests a number of tools to use in order to enhance knowledge and public awareness of nuclear science in general. The authors suggest communication and public outreaches and engaging media to share knowledge with the public. Public gatherings, road shows or public debates could be effective medium to discussing issues surrounding cancer and the how nuclear science can be useful in fighting the disease. In Malaysia, the ‘I love Nuclear’ Roadshow in 2010 was kick started with an aim to record basic data on public understanding. Similar shows can be deployed in different countries for a broader reach of the public. In the current times, television, the internet and social media are booming and could be very instrumental as information sharing platforms to reach the young and the old.

It is also important for academic institutions to include nuclear science into academic curricular and to establish medical and research institutions, as well as universities in order to develop professionals in nuclear science and radiology. Research centers and universities are a good platform for developing new knowledge, disseminating quality and reliable information. The International Atomic Energy Agency (IAEA) was particularly set up precisely to promote the good applications of nuclear energy in 1957 and it has done significantly a great job and paved the road for upcoming institutions with similar mandates. Lesotho is already moving towards establishment of a Cancer Centre for treatment and control of cancer in the country. This will be a great achievement for the country and will help ease the burden of foreign care and treatment which is both economically straining the country of its limited resources, and on the other hand putting emotional strain on family members who do not get to be near their patients. Particular care should also be taken when dealing with men. According to the Lesotho Demographic and Health Survey of 2014, when comparing knowledge of breast cancer in men and women with regard to the level of education, more women appear to know/or have heard about breast cancer than men in all levels of education. Naturally, men are not very comfortable in discussing health related issues and even taking action to seek medical care, hence special care should be taken to include men.

Conclusion

Most people associate the word “nuclear” primarily with weapons, and to a certain extend with nuclear power accidents. These associations often shadow in the beneficial uses of nuclear. There is need therefore, to enhance awareness on the good uses and safe applications of nuclear science and its applications. Health care also requires knowledge and educated professionals to properly administer and manage nuclear medicine and radiotherapy. Public outreaches, public lectures, road shows, media debates and discussions are recommended to share truths and benefits of the nuclear science.

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To my dear husband and my lovely kids, mommy took a lot of precious family time from but you never complained, instead you cheered me on; I am grateful.

To IYNC, I saw this opportunity very late, only 5 days to the extended dead line, I had nothing in my hands or head at the time but I knew I must take part. I grabbed the opportunity and in a day I submitted the abstract. I was skeptical about the idea but you gave me the chance, thank you for giving me a great opportunity, world class exposure and a chance to learn. I am thankful.

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Role of Training Infrastructure and International Cooperation in Internationalizaion of Nuclear Education and Training

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I. INTRODUCTION

The interest in nuclear industry from the young specialists thinking about their future careers is not as high as fifteen years ago and there are many of reasons for that – nuclear accident, vague plans for nuclear future in leading countries with large number of NPPs and sometimes even negative image of nuclear overall. Recent study shows that few millennials see nuclear as a top emissions-free energy source, or as part of our future energy sources [1].

In order to increase attractiveness of nuclear field globally it is necessary to educate broad audience of people, to communicate clearly with general public about the advantages of nuclear power has and to position nuclear as backbone option for sustainable development.

The approach proposed within this work is further expansion of international cooperation in terms of research, education and training though joint involvement of the capabilities of educational, research and industrial organizations involving in the process students and young researchers from all over the world.

Among new-comer countries that show interest in nuclear technologies, there is a demand for training courses on the use of software to justify the safety and technical parameters pre-selected nuclear technologies.

As an example, foreign partners showing interest in Russian reactor installations are also interested in training courses using Russian software, which is used to justify the safety and technical parameters of NPPs under construction.

The paper discusses new approach to internationalisation of nuclear research education and training based on experience of Rosatom Technical Academy in conducting training courses and bringing together key stakeholders from education, science and industry.

II. THE NEED IN NUCLEAR EDUCATION AND TRAINING WITH A PRACTICAL USE OF SIMULATION TOOLS AND EXPIRIENCE IN INTERNATIONAL TRAINING USING SIMULATIONAL TOOLS AND GLOBAL TRAINING INFRASTRUCTURE

Competences related to safety assessment of nuclear facilities is a key for making knowledgeable decisions in the design, licensing and operation of nuclear power plants

The core of nuclear educating forums could be international training courses, which are the complex of training modules containing practical exercises and theoretical training, e-learning and interactive sessions conducted all over the world and have duration from 36 up to 504 academic hours of training overall. Modular training program for professional training, continuing education and retraining on safety assessment of NPP conducted in English and in different formats: eLearning modules including simulation tool practice and face to face training.

The main tool for that is new kind of international courses with the mobility of young researches and students around the world and also with the use of domestic simulation tools and research infrastructure. As an example, the course could be composed of 8 modules of theoretical and practical training with overall duration – 504 hours (14 weeks):

1) Training module «Introduction to practical aspects of reactor technologies»;

2) Training module on regulatory base issues;

3) Training module on preliminary safety analysis report (Including PRA);

4) Course on review of simulation codes: neutronics and thermal hydraulics;

5) Practical session on analytical simulator of NPP;

6) Fellowships with the use of Russian and international simulation codes for safety assessment;

7) Training module on economics of nuclear power industry;

8) Practical fellowship at a universities' research facility dedicated to accident analysis.

Target audience for such a course are students, young specialists and personnel of national organizations (regulator, NEPIO, TSO etc).

A comprehensive training like the one suggested above inevitably involves an impact from nuclear companies representing different sides of the industry, science and education.

There are different formats, in which training could be conducted, so each module could be in format, for example: of workshop, scientific visit, regional training courses, fellowship programme, international schools. Rosatom Tech has an experience providing the services in the following formats and combinations of them, including bringing together experts from the different sides of the industry. The biggest challenge in organizing this kind of training is to attract side, for whom training is not a priority.

Particularly, at the Rosatom Tech, regional courses are held on an annual basis within the framework of the technical cooperation project of Rosatom State Corporation and the IAEA on infrastructure development and safety assessment of pressurized water reactors with the active use of Russian computational tools [2].

An important part of the course is a laboratory workshop, created to develop basic skills for calculating the basic processes occurring in reactor installations, to justify their safety using Russian software systems [3]. An interactive course at the Technical Academy of Rosatom is being introduced into a web portal that combines lectures, test assignments and a practical test. Access to the portal is open to all registered users.

The portal plays a role of an entry point and an invitation to attend a full-time course. Correspondence use of the course will give an overview of the technology and prepare the listener for a face-to-face meeting, and full-time training, in turn, will speed up the testing process, simplify the use of software for a practical cases, demonstrate the full range of uses of Russian calculation codes.

The proposed online course will include highly specialized courses on nuclear power and domestic technologies, read by leading experts of nuclear industry, as well as use special simulation codes for practical sessions.

Motivation for developing such a big program is a training of highly educated personnel with very specific knowledge, skills and competences which will be still required regardless of the development of nuclear power sector around the world, since new builds, development of innovative and advanced reactors, long-term operations, shut-down, decommissioning, waste management and radiation protection – all these sectors has a lack of qualified staff already or will face this lack in near term future.

Such training facilitates international cooperation between universities, industry and research institutes and will help to:

- maintain international cooperation between industry, universities and institutes to ensure high quality of education in accordance with international standards, keeping the level of teaching and preserving knowledge in the field of research, education and training in the field of nuclear science and technology;
- build joint educational programs to prepare a community of highly educated and experienced experts in the field of nuclear science and technology;
- provide an opportunity to exchange existing curricula, programs and handbooks for students, graduate students and trainers;
- joint development of new curricula, programs and handbooks;
- joint participation in international projects related to the development of nuclear research, education and training, including implementation of advanced simulation tools;
- improve the image of nuclear power as one of the key tools to fight climate change in the world;
- perform joint research to support enhancement and justification of modern pressurized water reactors safety features, as well as the development of fourth-generation reactor technologies, including commercial fast reactors to create a two-component nuclear power industry.

III. Conclusion

In order to increase attractiveness of nuclear field globally it is necessary to educate broad audience of people, to communicate clearly with general public about the nuclear power. The core of nuclear educating forums could be international training courses, which are the complex of training modules containing practical exercises and theoretical training.

Creating such a flexible international training tool for promoting nuclear technologies all around the world is challenge requiring effort of a young and passionate community of nuclear professionals. IYNC can serve a s global coordinator providing a framework accumulating an experience and best practices from companies, such as Rosatom Tech and its global partners.

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Overcoming Barriers on Communication,Education and Knowledge Management in Nuclear Technology in African Sociaties

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I. Introduction

Apart from promoting gender equity, the International young nuclear congress (IYNC) in line with its theme of next year's congress "**Diversity in Nuclear**" has given an opportunity to the third world countries to participate in such a great event in Sydney Australia. Communication, Education and Knowledge management are key in solving our global problems through Nuclear Science and Technology. We need first to communicate to the world so that the people know what nuclear science is and Technology, its benefits, advantages, disadvantages if there any and we have to be open about across all societies in the world. Then after communicating, we need to educate the people especially the youth of the world on nuclear science and technology and once we educate we talk about how we are going to manage the knowledge we have acquired and put it to right/good use in order to improve the world. Communication, I believe if everyone has the information on nuclear science and technology many challenges we are facing as the global can be reduced and almost completely eradicated starting from Health related problems, energy problems, agricultural improvement strategies, in manufacturing to mention but a few. In this regard it is important that there is global sensitization is put in place in schools, communities, villages and among the especially those in the sub-Saharan African countries and some parts of Asia. However we must use people in leadership to push our agenda in nuclear science and technology Education is key to any development. If people are informed, educated on nuclear science and technology even on how diverse it can be used and work then we will be home and dry. Most African countries don't have intuitions which offer such programs in nuclear science and if that can be introduced it can be great and a step further, here we are talking about diversity but most countries don't even know or having the knowhow to use nuclear science in any form known even the simple.

Moreover nuclear science and technology as program is not in most African intuitions hence if we intensify into training and education and diversify to any form it can best use in order to solve problems also teaching approaches should be put in place in order to have a broad understanding. Knowledge management. People have the negative approach in nuclear science they think it's only for weapons and anything evil relating to it, if we manage the knowledge we get from it nicely the better. The debate about nuclear is all over the world on the safety of its use but according to the world Nuclear Association (WNA) and Environmentalist for Nuclear Energy contend that nuclear power is safe, sustainable energy source and that it also reduces on carbon emissions in its operations. However, we can't rule out the need to research and develop towards greater efficiency, safety and other positive things

Furthermore I believe in countries where nuclear science and technology is still not taken serious or maybe it's due to lucky of resources are unable to venture into nuclear science and technology, hence i would suggest that these states can get into regional management in nuclear science and technology were they can set up nuclear energy plant for 3-5 countries with a common purpose to supply electricity to these countries and in Africa we find the energy deficit is growing day by day.

II. OBJECTIVE, RESULTS AND DISCUSSION

A. communication

The objective of this study is to discuss the experience of Communicating Overcoming barriers on Communication, Education and Knowledge management in Nuclear Technology in African societies.

Communication is a great privilege. It is an opportunity to spread knowledge by talking, writing or texting. It is also a serious responsibility, because others will benefit by the words. Therefore, it must be kept with all diligence and be filled with Excellent content to share with others. Because of the public association to the Hiroshima and Nagasaki bombings in the end of the World War II, as well as to the broadly known incidents of Three Mile Island, Chernobyl, Goiania and Fukushima, the nuclear sector has historically Encountered difficulties in communicating and bonding with Societies all over the world.

In order to effectively and efficiently communicate policies prior to the problems of nuclear communication issues, the question of what kind of message needs to be delivered, and to whom, is basically the most basic consideration There are many limitations in the concrete methodology that can increase the acceptance of nuclear power generation by local residents. We need to determine what factors influence acceptability, and how to set-up the interaction and the dynamic relationship between these factors.

B. Nuclear Education

Nuclear education allows for the effective use of educational resources valuable experiences and best practices, as well as education materials and tools can be channelled through meaningful regional and interregional cooperation.

A system of knowledge organization serves as a bridge between the user's need for information and the material with which it is counted, so they can be understood as schemes that facilitate the organization of information that produces knowledge. Efforts to establish a common language within the nuclear Community have been undertaken on international, national and organizational levels. Organizations like UNSCEAR, United Nations Scientific Committee on the Effects of Atomic Radiation; WHO - World Health Organization; ICRP - International Commission on Radiological Protection; or ICRU- International Commission on Radiation Units & Measurements, issue recommendations or safety standards Which, without having a binding character, give a common reference to national regulations.

According to the World Nuclear Association there is only one country which is in full Nuclear operational in Africa and others are coming up so there is need to increase education centres for this noble cause.

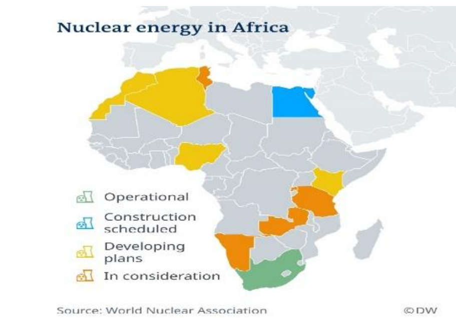

Figure 1. African Countries with Nuclear program.

However, Limited resources available for science research and education in African countries preclude most young scientists from attending international events such as conferences or international research internships. As a consequence, and despite attending prestigious educational programs, many African students find themselves at a systematic disadvantage; they do not get exposed to the same opportunities, experiences, or training, as their counterparts from more affluent institutions. This can cap the development of their communication and technical skills and keep them from reaching their full potential as scientists in an increasingly globalized society. This detriment can manifest as challenges in achieving English proficiency as the main language in scientific communications, limited interaction with scientists working on cutting-edge research, lack of Access to mentoring from such researchers, and restricted scientific and technical training for technologies not yet Available in their countries

C. Knowledge Management

The knowledge management within the Nuclear Engineering Department (DIN Spanish acronym) seeks to preserve and improve the operational capability of the department and it is developed through an integrated and systematic plan of human resources, processes, analysis tools, and regulatory aspects. The figure 2 below shows graphically the types of knowledge we can use to integration process of nuclear knowledge and figure 2 shows the knowledge management strategy. This management aims to identify, share, preserve, distribute, improve and expand the strategic and operational knowledge to be used in the processes of its concern, adding value to what we know.

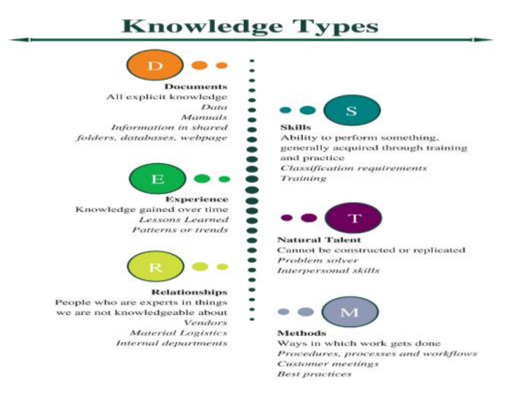

Figure 2. Knowledge Types

Figure 3.

IYNC2020
SYDNEY

III. Conclusions

However this can bring regional integration, for example if Zambia, Zimbabwe and Botswana get into this deal, Zambian can be supplying the Uranium, Zimbabwe can supply with other materials then the plant is set in Botswana this will make all the countries work hard in order for them to benefit equally In addition to the fact that nuclear science is becoming a hotcake I believe if nuclear knowledge is shared to the world, it would be better a place, in peace and able to solve different challenges we facing right now, hence communication, education and knowledge management are key to this.

Acknowledgment

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Virtual Reality Emergency Response Training at the IAEA

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I. Introduction

Preparation is essential for emergency responders to effectively and safely react to incidents involving radiological and/or nuclear materials; however, hands-on experience with significant quantities of these materials can often be implausible from a safety perspective or cost-prohibitive. The use of virtual reality (VR) tools allows emergency responders to participate in simulated responses using radiation detectors and other emergency response equipment to render a scene safe under the guidance of trained staff. The Incident and Emergency Centre (IEC) of the International Atomic Energy Agency (IAEA) has developed VR emergency response training systems using publicly-available hardware and software. This system has been used by responders to practice on-scene response actions, by public information officers to practice working in a simulated press briefing and by members of the public during public outreach events to show them a radiological emergency scene.

II. Development

A. Background

The IAEA IEC is the global focal point for international emergency preparedness, communication and response to nuclear and radiological incidents and emergencies, regardless of the triggering circumstances [1]. The IAEA has roles for emergency preparedness and response (EPR) covering response to incidents and emergencies and EPR preparedness activities such as developing Safety Standards, guidance and tools, by assisting IAEA Member States in the implementation of this guidance and the IAEA offers review services to Member States.

A recent and important IAEA training activity in EPR has been its School of Radiation Emergency Management (SREM), which has been conducted since 2015 [2]. This school is designed to strengthen national, regional and international capacity to respond to nuclear and radiological incidents and emergencies. The school consists of up to three weeks of training which includes expert lectures, interactive practical sessions, emergency response exercises and other content designed to teach the basic principles of EPR based on the IAEA Safety Standards and guidance.

Training and exercises are essential components for being prepared to respond to a radiological or nuclear incident or emergency. Practical, hands-on exercises can test skills, procedures, evaluate knowledge, and demonstrate responder capacities for the conduct of effective and safe response activities. Within nuclear and radiological emergency preparedness and response, many IAEA Member State national authorities as well as private institutions have well-developed emergency exercise programs covering the types of emergencies which may be possible at their facilities; however, even the best exercises have limitations: Emergency scenarios at nuclear power plants often rely on multiple, cascading failures to result in a release (which can be unrealistic); some exercises strive to end with positive outcomes (which may be unobtainable); and simplified or simulated response elements often have no practical consequences for the exercise participants to manage (which is unexpected). Less-than-ideal exercises can have real consequences for emergency responders, such as developing a false expectation of how an emergency response will take place, developing misconceptions of hazards or even bad habits due to the scenarios being “too safe,” and they may not be able to practice and evaluate the full scope of response procedures.

Staff in the IAEA IEC began developing the concept of leveraging newly-released consumer VR hardware in 2016 as a tool to support the IAEA EPR training program. During the exploration phase, the staff determined that it would be possible to develop synthetic computer-based emergency response scenarios with features such as accurate radiation fields and realistic hazards such as fire, smoke and electrical elements. These emergency response scenarios could then form the basis of emergency response exercise for IAEA EPR training activities. One of the goals of these efforts was to explore more interactive emergency response training activities where features such as the radiation fields could be much higher than traditionally possible during live radiological material training due to safety concerns.

B. *Technical Specifications*

The VR emergency response training system was developed by IEC staff using publicly-available hardware and software. In 2016 and continuing now, effective and realistic VR software required the investment of high-end Windows-based laptops or PCs. In order to meet the rigorous software demands, the IEC purchased "gaming" laptops with the following hardware specifications:

- Processor: Intel Core i7-7820HK @ 4 GHz
- Graphics Card: NVIDIA GeForce GTX 1080 (Laptop) - 8192 MB, Core: 1582 MHz, Memory: 2500 MHz, GDDR5X, ForceWare 375.63
- Memory: 32768 MB, 2x 16 GB DDR4-2400, Dual-Channel
- Display: 17.3 inch 16:9, 1920 x 1080 pixel 127 pixels per inch
- Mainboard: Intel CM238
- Storage: 2x Samsung SM951 MZVPV256HDGL (RAID 0), 512 GB SSDs + HGST Travelstar 7K1000 HTS721010A9E630, 1 TB HDD @ 7200 rpm

To support the VR experience for radiation emergency workers, the IEC purchased HTC Vive [3] VR systems with the following hardware specifications:

- Screen: Dual AMOLED 3.6'' diagonal
- Resolution: 1080 x 1200 pixels per eye (2160 x 1200 pixels combined)
- Refresh rate: 90 Hz
- Field of view:110 degrees
- Safety features: Chaperone play area boundaries and front-facing camera
- Sensors: SteamVR Tracking, G-sensor, gyroscope, proximity
- Connections: HDMI, USB 2.0, stereo 3.5 mm headphone jack, Power, Bluetooth

The IEC staff designed emergency response scenarios using the game engine Unity [4]. The Unity game engine is a software product which brings together elements such as graphics, audio, networking, physics, and scripting capabilities into a single package for users to create their own games without the need to develop these features independently [5]. In addition, Unity provides support for a number of different platforms which include Windows-based PCs, home game consoles and mobile devices (such as Android and iOS). In addition to Unity, the OpenVR API [6] was used with the SteamVR unity asset [7] for developing the VR interface and interactions.

III. Emergency Response Training in Virtual Reality

A. *Set-Up*

The HTC Vive headset must be connected to the computer with the HDMI, USB 2.0, stereo 3.5 mm audio, and power cables. In addition, two base station units must be positioned at selected locations around the desired "play area." The base stations emit infrared signals that are detected by the HTC Vive headset in order to establish a "room scale" area, where the user can walk around, turn and/or look in different directions, and have his or her movement be reflected in virtual reality. The base stations can be up to 16 feet apart and are best mounted on tripods or fixed to walls, in order to ensure sufficient line-of-sight communication with each other and the headset.

After powering-on the base stations and laptop, and opening the virtual reality simulations, the user must activate the two wireless HTC Vive hand-held controllers (often called wands) and establish a safe "play area" during setup and for interacting with the world in virtual reality within the SteamVR chaperone system. A sample set-up is shown in Figure 1 below:

Figure 1. HTC Vive set up, with one base station visible in the background (Photo: Y. Yustantiana/IAEA)

B. *Emergency Response Simulations*

The IEC has actively developed several different scenarios for VR emergency response training. These scenarios include aspects such as:

- Searching for radiological sources during a traffic accident
- Recovering radiological sources from a transport accident at a port
- Determining and evaluating hazards during a train derailment with spent nuclear fuel
- Consequence assessment and recovery operations following a spill of material at a nuclear power plant site

C. *Interactions in Virtual Reality*

A defining characteristic of the IEC virtual reality emergency response training is the realistic behavior the radiation physics. IEC staff created digital radiation survey meters that users can pick up, move around, and read their display in virtual reality. As users approach radioactive material in VR they will see a realistic response on the detector accounting for the decreased distance to the radiation source. These calculations conducted an inverse square law calculation for each radiation source to determine the resultant exposure to the radiation detector at each given moment. In addition, the IEC staff created different materials in the environment, such as building materials, shipping containers, metals, plastics, and wood. When placed between the user's radiation survey meter and a radiological source, these materials realistically reduce the measured dose rate using half value layer attenuation coefficients from a library [8]. In this way, users participating in the VR emergency response training are able to apply their training on health physics and radiation safety to account for shielding as they conduct their emergency response procedures. In addition, the IEC included basic tools such as a flashlight and a telescope for users to pick up and use in virtual reality.

D. *Movement in Virtual Reality*

As a "room scale" virtual reality experience, the HTC Vive allows users to walk around within the "play area" and their movement will be reflected in the simulation. Turning and looking in different directions will also be reflected in the simulation. However, simulations with larger areas such as outdoor emergency scenarios cannot be recreated at a 1:1 scale in an office building or classroom. Therefore, users can digitally move within the virtual reality simulation with one of two methods: Firstly, users can use their thumb to push on a direction-pad located on one of the wireless hand-held controllers. This will enable the user to "walk" or "fly" in the direction they push. Secondly, users can hold a button on one of the wireless hand-held controllers to crate a "teleport" effect – the user will automatically appear at the location they are pointing to with their wireless hand-held controller when they release the "teleport" button. These two options enable users to cover large and small areas in virtual reality without leaving the defined "play area," and was important for the IEC staff to be able to program emergency response scenarios in large, outdoor areas as shown in Figure 2, below:

Figure 2. User in an "outdoors" virtual reality emergency response scenario while "holding" a radiation survey meter (Photo: Y. Yustantiana/IAEA)

E. *Training and Exercise Conduct*

Whether for demonstration or training and exercise purposes, IEC staff always conduct virtual reality emergency response training with an instructor or proctor. This is important for several reasons: Firstly, the participating user is wearing a headset that obscures their vision of their surroundings in real life. Although the SteamVR chaperone system has safety features built-in to alert the user if they approach the edge of the safe "play area" zone, a proctors first responsibility is to ensure the safety of the participant. Secondly, the value of the emergency response scenario is not gained just from a participating user looking or walking around in a virtual environment. The value is gained from the user answering questions about the hazards he or she can identify, providing instant feedback to the guide or proctor about their observations, and from other external observers learning from the actions the participating user takes in virtual reality. The IEC staff developed standard checklists of questions for proctors to ask users who participate in each of the scenarios listed above.

IV. BENEFITS OF VIRTUAL REALITY EMERGENCY RESPONSE TRAINING

A. *Advantages*

There are several advantages to the use of virtual reality for emergency response training. New technology enhances classroom interaction and engagement, and instructors or proctors can provide immediate feedback or guidance to reinforce key concepts that are being taught. Because the virtual reality system can be connected to an external display, the instructor or other learners can see what the participating user sees, and often this allows for immediate corrective action advice as well as a better understanding for 'why' an incorrect action may be taken by the user.

Virtual reality emergency response training allows for the creation of complex and/or high-risk scenarios for users to exercise live. These scenarios may take place in very large-scale environments (such as the previously-mentioned outdoors scenarios) by expanding the scope of a classroom drill, and real equipment such as automobiles, trains, trucks, aircraft, and more do not need to be taken out of service for extended periods of time to conduct a field exercise in real life. Similarly, real-life locations that are generally impractical for conducting emergency response exercises (such as ports, airports, city streets, highways, and more) can be realistically recreated in VR.

Finally, different countries and emergency response programs have access to different facilities and locations; but, in virtual reality any user can participate in a scenario regardless of whether the specific type of location or facility exists in their country. This can enable radiation emergency workers in land-locked countries to practice responding to events at shipping ports, allow responders from countries without a nuclear power plant to experience a simulated emergency at such a plant, or provide an opportunity for public information officers to feel pressure of answering questions in front of a room of virtual reporters.

B. Uses of Virtual Reality

Between 2017 and 2019, the IEC staff has utilized virtual reality emergency response training for the following meetings, symposia, and schools:

TABLE I. USES OF VIRTUAL REALITY

Date	*Topic*
October 2017	IAEA Technical Meeting on Response to Incidents or Emergencies Involving Transport of Nuclear or Radioactive Material
November 2017	IAEA Workshop on Assessment and Prognosis to a Nuclear or Radiological Incident or Emergency
April 2018	IAEA involvement in Austria's Long Night of Research
September 2018	62nd Annual Regular Session of the IAEA General Conference
October 2018	International Symposium on Communicating Nuclear and Radiological Emergencies to the Public
November 2018	IAEA School of Radiation Emergency Management
December 2018	IAEA Conference on Nuclear Security
October 2019	IAEA School of Radiation Emergency Management

Each meeting or topic consisted of a dedicated workspace where IEC staff set-up the virtual reality equipment and guided participating users through one or more scenarios. In the Technical Meetings and Workshops, participants who were not actively using the virtual reality system watched others and provided feedback, including instructions for personal safety and technical guidance for approaching suspected radioactive materials.

C. Lessons Learned

Adult learning principles require a realistic play environment and natural gestures to be most effective. Put simply, users participating in virtual reality training need to feel like they are actually "there." Past negative experiences from some participants using earlier virtual reality models left some participants with poor views of such digital experiences; however, once users were given an opportunity to use the system on the HTC Vive their opinions frequently changed.

Participants are able to understand how to interact in an environment and move around in virtual reality with very short training sessions – typically three minutes or less – regardless of their age or comfort with technology. The guidance of an instructor or proctor is an important element of this comfort, in addition to being an essential component of a focused training experience.

Classroom discussions or exercises can be made considerably more appealing with the use of virtual reality, and participants have provided feedback that they learn better from realistic scenarios where the hazards are visible and failure can result in a premature end of the scenario (for example, if a participating user responding to an automobile accident gets struck by a car because they did not look for traffic). Overall, feedback has been overwhelmingly positive, and the IEC staff will look to expand the use of virtual reality emergency response training.

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Critical Success Factors for Implementing of Knowledge Management in Nuclear Organizations: A Fuzzy Approach

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I. Introduction

The knowledge of workers constitute as valuable resources, as they enable organizations to perform their functions successfully. However, there are conditions that favor the loss of this knowledge in organizations, as for example, the natural aging of workers and consequently the retirement and staff turnover. Then, it becomes important for organization to seek the preservation these knowledge. For a successful implementation of Knowledge Management (KM), it is important to identify the barriers or critical factors that affect the success of the KM process.

From the perspective of the nuclear organizations, no systematic framework exists on characterizing a set of critical success factors (CSFs) for implementing KM. Furthermore, the CSFs assessment deals with uncertainty and imprecision of human judgments. In this context, this paper presents a set of CSFs and a fuzzy model to establish a standard of importance of these CSFs based on experts opinion for the implementation of KM in nuclear organizations. Fuzzy theory is essentially used in mapping quantitative models for decision making and representation methods in imprecise and uncertain environments.

II. Critical Success Factors

The set of critical success factors can act as a list of items for organizations to address when adopting knowledge management. This helps to ensure that the essential issues and factors are covered during design and implementation phase. For academics, it provides a common language for them to discuss and study the factors crucial for the success of knowledge management program in an organization.First, confirm that you have the correct template for your paper size. This template has been tailored for output on the US-letter paper size.

However, no systematic work exists on characterizinga collective set of CSFs for implementing KM in nuclear organizations. CSFs are critical areas of managerial planning and action that must be practicedin order to achieve effectiveness. In terms of KM, CSFs can be viewed as those activities and practices that should be addressed in order to ensure its successfulimplementation. These practices would either need to be nurtured if they already existed orbe developed if they were still not in place [1].

III. Basics of Logic Fuzzy

Fuzzy logic provides an appropriate logical-mathematical framework to handle problems with such characteristics [9], since: (1) it deals with uncertainty and imprecision of reasoning processes and situations; (2) it allows the modeling of the heuristic knowledge that cannot be described by traditional mathematical equations and; (3) it allows the computation of linguistic information.

Several studies show important reasons to use fuzzy set theory (FST) [2][3][4]: reduction of human error, creation of expert knowledge and interpretation of large amount of vague data.

Fuzzy set theory (FST) is an extension of classical set theory where elements have degrees of membership. Let X be the universe of discourse and x a generic element of X, a fuzzy subset $\tilde{A}$, defined in X, is one set of the dual pairs, as in (1):

$$\tilde{A} = \{(x, \mu_{\tilde{A}}(x)) \mid x \in X\} \qquad (1)$$

where$\mu_{\tilde{A}}(x)$ is the membership function or membership grade x in A. The membership function associates to each element x of X, a real number $\mu_{\tilde{A}}(x)$, in the interval [0, 1].

IV. Method

The method developed in this paper was structured according to the following steps:

1) Selection of CSFs;

2) Prioritization of CSFs;

1) Selection of CSFs

The list of CSFs was developed in seven themes, based on the literature[1][5][6][7][8][9]: top-level commitment, organizational culture, organizational structures, human resources management practices and policies, measuring and results, information technology and learning culture.

The themes and CSFs are described in table I.

TABLE I. THEMES AND CSFS

Themes	Critical Success Factors
1- Top-level commitment	1.1 Mission and values: There is a clear definition of the mission and values of the institution.
	1.2 Goals and Objectives: The goals and objectives of the institution are clearly defined.
	1.3 Interaction between people: There is freedom of interaction between people and working groups.
	1.4 Knowledge sharing: Managers provide support for sharing knowledge betweenthe groups.
	1.5 Knowledge dissemination: The managers motivate and create internal conditions for the dissemination of knowledge.
	1.6 Investments to knowledge dissemination: There are investments for knowledge dissemination in the work sector.
2. Organizational culture	2.1 Organizational climate: There is a positive environment, encouraging the knowledge sharing.
	2.2 Incentive Program: There is an incentive for the ideas used by the institution.
	2.3 Meetings: People realized meetings to disseminate successful practices.
	2.4 Documentation: There is a practice of elaboration and disseminationof project/research documents reporting positive and negative aspects.
	2.5Relationship of Trust: There is a relationship of trust between people.
	2.6 Commitment: The workers are committed to the institution.
	2.7 Learning: The institution has an internal culture aligned to the learning process, that is, there is a learning environment in the institution.
	2.8 Positive environment: There is a positive environment for communication, cooperation and negotiation.
3. OrganizationalStructures	3.1 Multidisciplinary groups: The groups are multidisciplinary and have autonomy and decision action.
	3.2 Project/Research groups: Groups are organized by projects/research (processes).
	3.3 Cooperation between groups: There is cooperation between physically distributed groups.
	3.4 Formation of groups: There is freedom in the formation of groups.
	3.5 Administrative structure: There is a specific group responsible for knowledge management.
4. Human resources management practices and policies	4.1 Training: Appropriate training for activities is often offered and encouraged by managers.
	4.2 Time pressure: There is no pressure for time and excessive goals, which are barriers to knowledge.
	4.3 Additional efforts: Additional efforts related to knowledge and performance are recognized by the leadership.
	4.4. Identification of competencies: There is an appropriate procedure for identifying skills and selecting people to work on the site.
	4.5. Human Resources: The capacity of workers in the sector is sufficient to ensure the sharing of information.
	4.6 Responsibilities: The responsibilities attributed to persons are accompanied by criteria for the valuation.
5. Measurement of results	5.1 Investment monitoring: There is a monitoring of the results of investments in training.
	5.2 Performance assessment: There is an evaluation of the

	results (performance) of activities of the workers.
	5.3 Alignment identification: The activities developed by the people are evaluated regarding the alignment with the objectives of the institution.
6. Information Technology	6.1 Technological structure: There is a technological structure for the storage and sharing of knowledge.
	6.2 Information access policy: There is a portal (local network)of dissemination and search for knowledge.
	6.3 Training: People are trained to use the technological resources of information.
	6.4 Availability of information technology: The technological resources of information are available to people when needed.
	6.5 Quality of technological resources: The information technology resources are efficient for storing and disseminating information.
	6.6. Access information: The information is centralized in a system of easy access.
7. Learning culture	7.1. Context changes: People are encouraged to learn, participate and accept new practices and innovative technologies.
	7.2 Contents of the Documentation: The procedures, instructions or documentation are updated and easy to understand.
	7.3 Information content: The information exchanged during the communication processes is sufficient.
	7.4 Communication: Communication mechanisms are efficient for disseminating information about work activities (knowledge).

2) *Priorization of CSFs*

The second step of this fuzzy framework is to obtain from experts on KM the degree of importance of each CSF, so that the implementation of KM in organization can be considered good. This phase has the following steps:

Calculation of experts' relative importance. The relative importance of the expert will be calculated on the basis of experts' attributes (experience, knowledge of KM). We will use a questionnaire (Q) to identify the profile. Each questionnaire will contain information of a single expert. The relative importance (RI) of expert Ex_i ($i = 1, 2, 3,\dots, n$) will be a subset $\mu_i (k) \in [0,1]$ defined by Eq.3. Referring to Eq.3, *tQi*, will be the total score of expert *i*.

$$RI_i = \frac{tQ_i}{\sum_{i=1}^{n} tQ_i} \quad (3)$$

Choice of linguistic terms and membership functions. Each CSF can be seen as a linguistic variable, related to a linguistic terms set associated with membership functions. These linguistic terms will be represented by triangular fuzzy numbers to represent the importance degree of each CSF (Fig. 1). It will be suggested that the experts employ the linguistic terms, U (Unimportant), LI (Little Important), I (Important) and VI (Very Important) to evaluate the importance of each CSF.

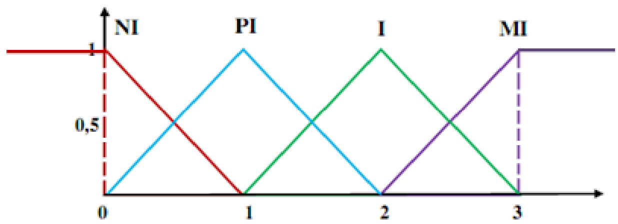

Figure 1. Membership functions

Aggregation of the fuzzy opinions. The similarity aggregation method proposed by Hsu and Chen [10] will be used to combine the experts' opinions which are represented by triangular fuzzy numbers. The agreement degree (A) between expert Exi and expert Exj will be determined by the proportion of intersection area to total area of the membership functions. The agreement degree (A) is defined by Eq.4.

$$A = \frac{\int_x \left(\min\left\{\mu_{\tilde{N}i}(x), \mu_{\tilde{N}j}(x)\right\}\right)dx}{\int_x \left(\max\left\{\mu_{\tilde{N}i}(x), \mu_{\tilde{N}j}(x)\right\}\right)dx} \quad (4)$$

If two experts have the same estimates, then, A=1. In this case, the two experts' estimates are consistent, and then the agreement degree between them will be one. If two experts have completely different estimates, the agreement degree will be zero. If the initial estimates of some experts have no intersection, then we will use the Delphi method to adjust the opinion of the experts and to get the common intersection at a fixed α – level cut [14]. The higher the percentage of overlap, the higher the agreement degree. After all the agreement degrees between the experts calculated, we will construct an agreement matrix (AM), which will give us insight into the agreement between the experts.

$$AM = \begin{bmatrix} 1 & A_{12} & \cdots & A_{1j} & \cdots & A_{1n} \\ \vdots & \vdots & & \vdots & \vdots & \\ A_{i1} & A_{i2} & \cdots & A_{ij} & \cdots & A_{in} \\ \vdots & \vdots & & \vdots & \vdots & \\ A_{n1} & A_{n2} & \cdots & A_{nj} & \cdots & 1 \end{bmatrix} \quad (5)$$

The relative agreement (RA) of expert Ex_i (i = 1, 2, 3, …, n) will be given by Eq.6.

$$RA_i = \sqrt{\frac{1}{n-1} \cdot \sum_{j=1}^{n} (A_{ij})^2} \quad (6)$$

Then we will calculate the relative agreement degree (RAD) of expert Ex_i (i = 1, 2, 3, …, n) by Eq.7 and the consensus coefficient (C) of expert Ex_i (i = 1, 2, 3, …, n) by Eq.8.

$$RAD_i = \frac{RA_i}{\sum_{i=1}^{n} RA_i} \quad (7)$$

$$C_i = \frac{RAD_i \cdot RI_i}{\sum_{i=1}^{n} (RAD_i \cdot RI_i)} \quad (8)$$

Let Ñ be a fuzzy number for combining expert's opinions. Ñ is the fuzzy value of each leading indicator which is also triangular fuzzy number. By definition of the consensus coefficient (C) of expert Ex_i (i = 1, 2, 3, …, n), $\tilde{N}$ can be defined by Eq.9. Referring to Eq.9, $\tilde{n}_i$, is the triangular fuzzy number relating to the linguistic terms, U (Unimportant), LI (Little Important), I (Important) and VI (Very Important).

$$\tilde{N} = \sum_{i=1}^{n} (C_i \cdot \tilde{n}_i) \quad (9)$$

Priorization. The priorization of CFFs will be established by calculating the normalized importance degree (NID) of each CSF. The normalized importance degree (NID) of each CSF will be given by deffuzification of its triangular fuzzy number $\tilde{N}$ (a_i, b_i, c_i), where b_i represents the importance degree. Then, NID will be defined by Eq.10.

$$NID_i = \frac{NID_i}{\text{the largest numerical value of bi}} \quad (10)$$

V. Conclusion

In this paper we described a method that uses CSFs and properties of fuzzy fets theory. We develop a priorization of CSFs using a similarity aggregation method to aggregate fuzzy individual opinions, considering the difference of importance of each expert. This prioritization is important to evaluate the CSFs that will influence the efficiency of the implementationof KM in nuclear organization. This means that this fuzzy approach is a proactive tool to provide a basis for checking the CFS in organization. As the result, the organizations that use this method will be able to proactively evaluate and manage knowledge.

As suggestions for future research, we highlight: (1) a pilot study using this fuzzy approach; (2) the development of a computational system in order to automate the use of the method to evaluate the CSFs online.

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IYNC2020 SYDNEY

The Influence of National Nuclear Youth Community (KOMMUN) on Indonesian Youth's Knowledge About Nuclear Science and Technology

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I. INTRODUCTION

Youth the most valuable asset for a nation. According to the Indonesian republic Act No. 40 of 2009 on youth in article 1, paragraph 1 that youth is Indonesian nationals entering an important period of growth and development over the age of 16 (sixteen) to 30 (thirty) years.

Rapid development of science and more advanced, such as technological tools. Emerging technology causes people increasingly easy access to all forms of information. One of them is the Nuclear Technology. Nuclear technology is one technology that is often experienced ups and downs, because they still imagined with incident dark period so it is very difficult to be accepted by the community, especially in Indonesia. For those trying to build atomic energy as power source, memory of Hiroshima and Nagasaki never far behind. As a result, nuclear technology categorizes something other than the atomic bomb that involved significant work (Garud, R. 2010).

In organizing, youth is seen as an agent of change (Warren, 2008). Associated with the agents of change, certainly has a great influence on public life. According to Christen, BD, and Dolan, T. (2011) that the organizers of the youth are the process that brings young people together to discuss the most pressing problems in their communities, researching on this issue and possible solutions and follow up with social action to create change at the community level.

As community of youth who engaged in the nuclear field certainly has challenges in educating Nuclear Science and Technology, see that the lack of public acceptance and insight on nuclear development in Indonesia. Is related to this, of course KOMMUN must have the knowledge and the communication of science to create a work and innovation in delivering and resolve this issue. Communication science emerged as the views concerning communication between the scientific community, interest groups, policymakers, and various public. KOMMUN expected to improve the knowledge of Indonesian youth in development of nuclear science and technology. Very important role of youth in the development process. As agent of change, youth have positive thoughts, spirit of struggle and be able to compete with the outside world.

II. KOMMUN AND KNOWLEDGE YOUTH

As youth community who engaged in nuclear science and technology development would have a major impact in changing environment. This change in view is based on, how participation of Indonesia in advancing the nation's youth through knowledge that has been received. Science is growing rapidly to make youth become increasingly develop innovative thinking in the development of technology. National nuclear youth community (KOMMUN) also have their opinion way of enhancing innovation and keep up with technological developments. Youth who engaged in nuclear technology has a goal to provide influence for the young man himself or society. Youth participation is a process that involves youth people institutions and decisions that affect their lives (Checkoway2006). Through participation of member KOMMUN it will be easy nuclear eduacating in community environment.

Before KOMMUN formed youth do not understand the benefits of nuclear, because they still imagined with incident dark days of World War 2 related to the outbreak nuclear bomb on the Japanese city Hiroshima. With the KOMMUN then able to change image of the Indonesian youth that nuclear is not about bomb but has benefits for life. Some activities KOMMUN giving great influence youth-related knowledge, including:

a. *Bincang-bincang Nuclear (BBN)*
b. *Nuclear Youth Summit (NYS)*
c. *Nuclear Talks (NT)*
d. *Nuclear Education Training (NET)*
e. *Nuclear on Weekend (NOW)*

The activity aims to provide members and general public with understanding and knowledge of nuclear.

III. METHODOLOGY

Through the activities carried on by KOMMUN can affect the way youth think about nuclear. Submission of nuclear packed simply so that it's understanding easily accepted by society. Youth who are agents of change will have many varieties of creative innovation in developing a knowledge.

The method used in writing is the quantitative method. Through this method, the author describes the data members to obtain a description of the number of members KOMMUN per year. Sources of data in the form of secondary data are data that has been collected is used to solve problems in the face (Sugiyono, 2013). Then the data is calculated based on the membership that has been registered. Here the number of members KOMMUN from 2013 to 2019.

TABLE I. NUMBER OF MEMBERS

No.	Years	membership
1	2013	72
2	2014	59
3	2015	135
4	2016	192
5	2017	96
6	2018	50
7	2019	45
Total		649

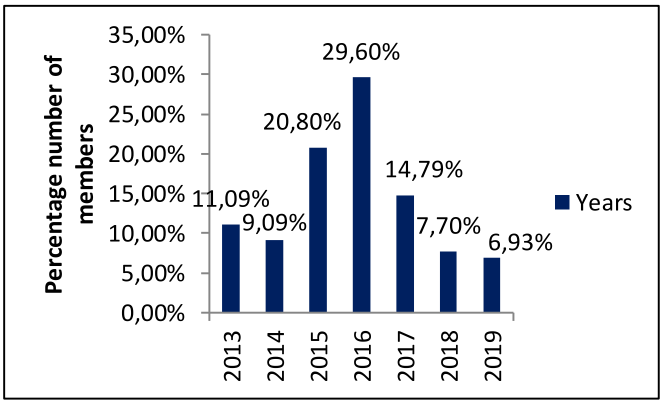

Figure 1. Percentage Number of members KOMMUN

Based on Figure 1 that the percentage of the sheer number of members KOMMUN most was in 2016 and the least was in 2019. By 2016 KOMMUN members. On years 2019 number of members at least it's still in open recruitment. In 2013 the number of registered members as much as 11.09%, of the total number of registered members, in 2014 as much as 9.09% registered members, 2015. 20.80% of registered members, in 2016 the number of members of 29.60%, 2017 the number of members of 14.80%, 7.70% in 2018 and by 2019 as much as 6.92%. The number of members from year to year is decreased and the activity of members on the wane. This is due to the lack of participation of the members of each activity undertaken. 2019 each KOMMUN region is still in the process of membership generation or open recruitment of members.

TABLE II. KOMMUN ACTIVITY

No	Years	Activity Name					
		BBN	NYS	NT	NET	NGS	NOW
1	2013	0	1	0	0	0	0
2	2014	0	1	0	0	0	0
3	2015	1	1	1	0	1	0
4	2016	10	0	7	2	2	0
5	2017	12	0	2	2	4	8
6	2018	9	0	2	5	1	1
7	2019	3	1	3	3	2	2
Total		34	4	14	12	9	11

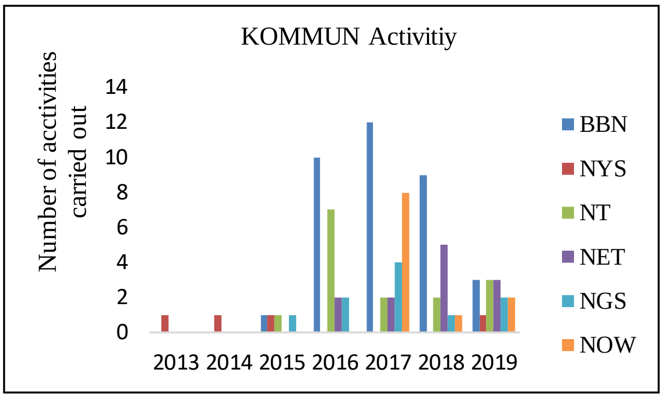

Figure 2. Number of KOMMUN activity

From picture above most effective activities nuclear talking (BBN). BBN is a relaxed discussion that talks about nuclear with a different theme. BBN aims to provide insight to KOMMUN members, youth and the general public related to nuclear. KOMMUN most productive activities, namely in 2017, for every activity carried out well. From 2013 to 2014 KOMMUN still in the stage of seeking membership generations so that only the activities Nuclear Youth Summit (NYS). This activity is carried out with the aim of collecting member, provides an understanding of nuclear KOMMUN and to each member of the region.

Starting in 2013 until 2019 KOMMUN contribute in providing an understanding of the nuclear environment. Since 2013 to 2019 member KOMMUN give Indonesia a major influence on youth-related to their rapidly developing nuclear power. Following grafic is number of people exposed to nuclear information from KOMMUN activity.

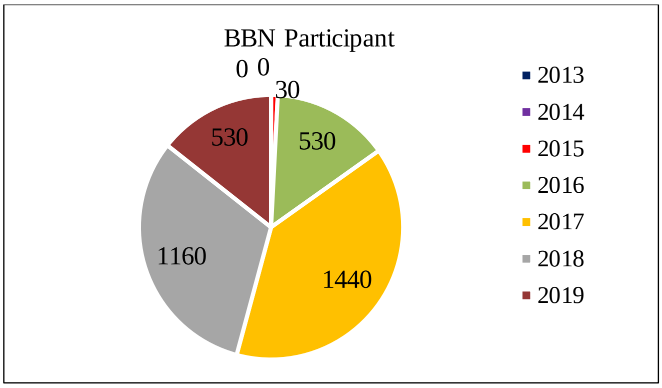

Figure 3. BBN Participant

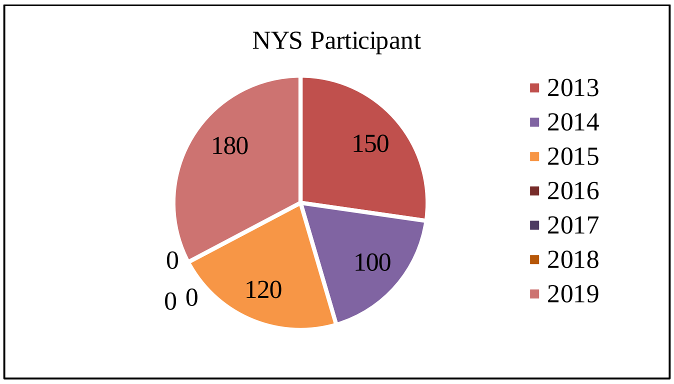

Figure 4. NYS Participant

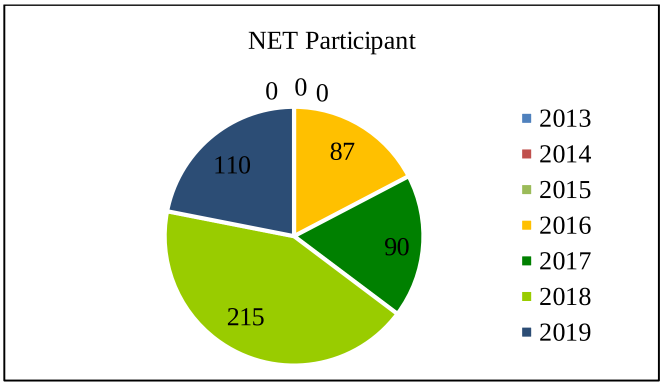

Figure 5. NET Participant

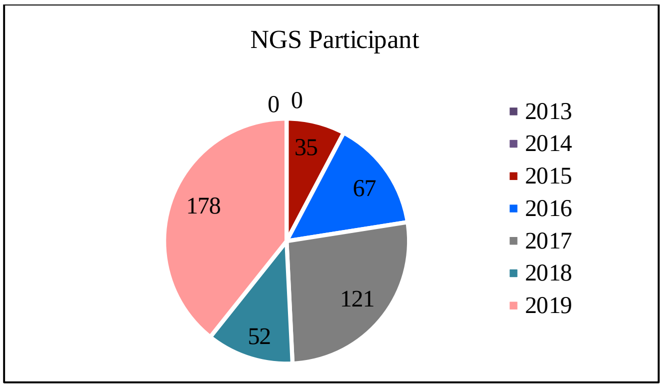

Figure 6. NGS Participant

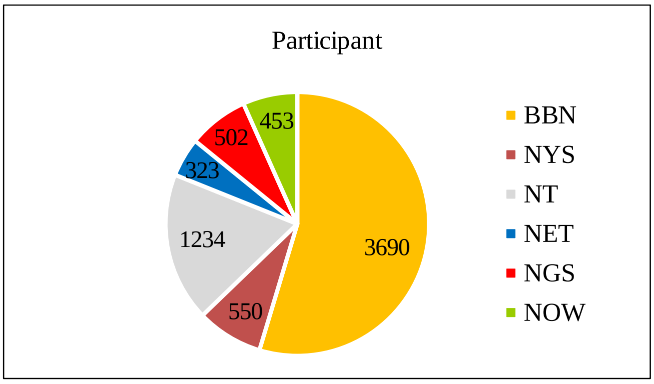

Figure 7. Participant activity

Data obtained from a range of regional activities that have been carried out. Based on the picture above that most participant activity is BBN. This activity really caught their attention, because BBN was delivered through online media so that information about nuclear was very easily accepted by public.

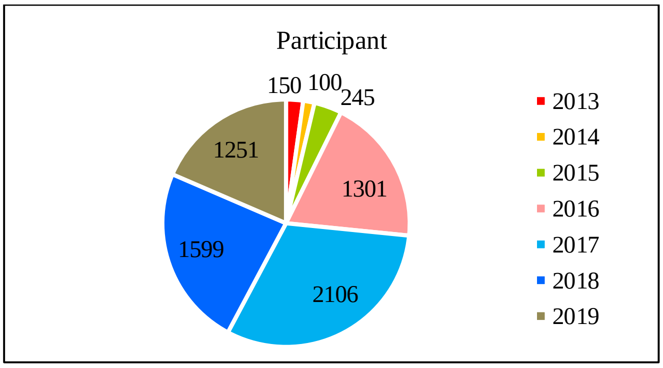

Figure 8. Participant every years

Data above shows the number of people received nuclear information each year based on the activities carried out by

KOMMUN. Figure 8 that 2017 is the most productive year and most number people receive nuclear information. From 2013 to 2015 there were still fes people received nuclear information, because KOMMUN is still in the development stage of membership.

IV. CONCLUSION

Based on the results of data analysis, it can be concluded that most effective KOMMUN activities is BBN and most productive year in 2017. It can be concluded that, BBN is the most accessible activity for public, so that can receive information about nuclear.This is seen based on the number of people participating in KOMMUN activities, so they are exposed to information about nuclear. KOMMUN activites certainly have a big influence on the development of nuclear science and thecnology in Indonesia, who initially only knew nuclear about bombs and through this activity they learned that nuclear is very beneficial for life.

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New Communication Campaign by the Spanish ENS-YGN: Basic Course of Medical Applications of Nuclear Technology

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I. INTRODUCTION

Nuclear technology is usually associated with energy generation, but this is only one of its applications. Nuclear technology is present during our day-to-day with many applications in several fields (i.e., in agriculture, food, industry, aerospace, art, among others), being the medical ones probably the most relevant. For example, among those cutting-edge applications, there are advanced radiodiagnostic technics and different therapy treatments based on nuclear technology. The first group of applications encompasses the medical tests that allow screening the human body, while the second group includes the direct applications of nuclear technology to cure diseases or tumors, pursuing to increase the life expectancy and the patient's healing.

It is widely perceived that people are not aware of the importance of the medical applications of nuclear technology for our lives. Consequently, the Spanish ENS-YGN Jóvenes Nucleares [1] has developed the new formative activity 'Basic Course of Medical Applications of Nuclear Technology' which objective is to bring students to the most important aspects of medical applications based on nuclear technology used routinely in the hospital's environment.

The course is divided into eight topics covering the two main fields of nuclear technology applications in healthcare, nuclear medicine and medical physics. The contents describe the main techniques for diagnostic purposes, such as Angiography, Computed Tomography (CT), Positron Emission Tomography (PET), Single Photon Emission Computed Tomography (SPECT) or Nuclear Magnetic Resonance (NMR), even though the last one is based on the hydrogen atoms resonance instead of ionizing radiation. Therapy treatments based on nuclear technology such as Radiotherapy (Teletherapy and Brachytherapy) or Nuclear Medicine are also treated along the topics. Finally, waste management and the latest advances in therapy technologies complete the content of the course.

Jóvenes Nucleares offered this course to physiotherapy and nursing students, external young professionals, and vocational training students in the healthcare field, trying to bring students closer to the medical applications of nuclear technology to see how they can help healthcare professionals saving lives.

II. CONCEPTION

As said in the previous section, the first edition of the course was mainly addressed to physiotherapy and nursing students, external young professionals, and vocational training students in the healthcare field.

From Jóvenes Nucleares we realize that there are some professional profiles that don't receive education about what is radioactivity, how is it used in hospital, how are the radioisotopes produced. It is true that these professional profiles don't manage directly this kind of technology (nuclear applications) but it is possible that their patients do receive treatments based on nuclear technology.

At the same time, we see a great demand of information on medical applications of nuclear technology. In this course there were many questions from the attendees that were curious about this field.

The proposed topics were the following:

Topic 1: Introduction to Medical Physics [2-3].

In this first chapter we try to answer the following questions:

- What is radioactivity? Structure of an unstable atom. Fundamentals of radioactivity. Radioactivity History.
- What is radiation? Types of Radiation. Interaction of radiation with matter. Radiation exposure.
- What is Medical Physics? Introduction. Applications of ionizing radiation. Medical Physics in Spain.

Topic 2: Radiodiagnostic equipment [4].

In this topic we speak about:

- X-ray production.
- Computed tomography (CT).
- Mammography.
- Angiography.
- Nuclear Magnetic Resonance (MRI).

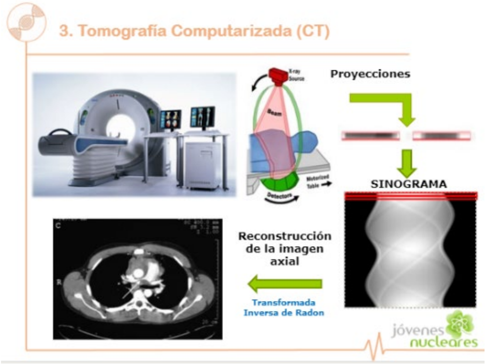

Figure 1. A slide from Topic 2.

Topic 3: Radiotherapy [5-6].

This theme covers the two type of radiotherapy, the remote radiotherapy and the internal radiotherapy (brachytherapy). We also speak about the planning system used in hospitals for this technology.

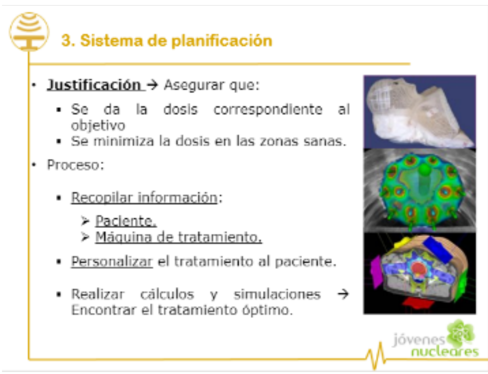

Figure 2. A slide from Topic 3.

Topic 4: Nuclear medicine [7-8-9].

In this chapter many important concepts are discussed:

- What is nuclear medicine?
- Radiation detection systems in Nuclear Medicine.
- Gamma camera.
- Single photon emission computed tomography (SPECT).
- Positron emission tomography (PET).
- Multimodal systems: SPECT/CT and PET/CT.
- Production of radionuclides and radiopharmaceuticals.
- Internal dosimetry.
- Waste management.

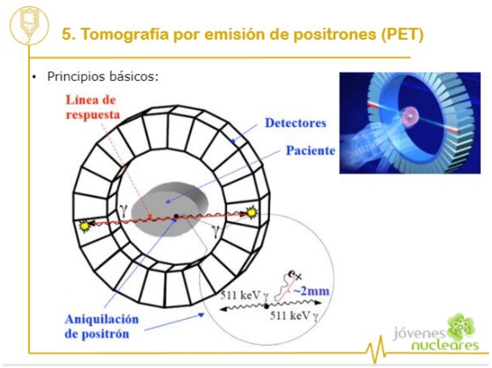

Figure 3. A slide from Topic 4.

Topic 5: Radiation protection [10-11].

This subject is dedicated to explaining:

- How ionizing radiation affect us and its biological effects.
- How do we measure them?
- Dosimetry Magnitudes, units and measuring devices.
- How to protect ourselves. ALARA principle and dose limits.

Figure 4. A slide from Topic 5.

Topic 6: Development and future [12-13-14-15-16-17].

In the last chapter we speak about:

- Advances in Medical Physics.
- New Medical Linear Accelerators.
- Proton therapy.
- Boron Neutron Capture Therapy (BNCT).
- Planning with Monte Carlo.
- Improvement of internal dosimetry models.
- Dose registration (2013/59 / EURATOM).

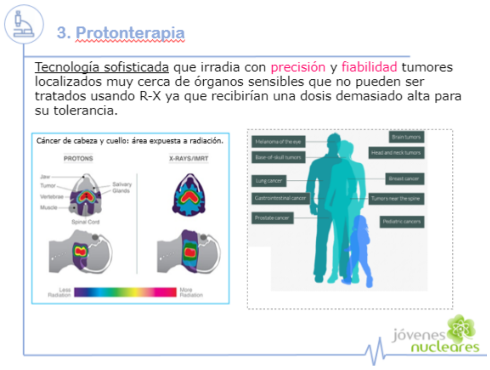

Figure 5. A slide from Topic 6.

III. First Edition of the Course and Conclusion

Jóvenes Nucleares offered the first edition of the Basic Course on Medical Applications of Nuclear Technology [16] in October 2018 at the Catholic University of Ávila (UCAV) where 200 young people attended the course. In that first edition, the course was mainly addressed to physiotherapy and nursing students, external young professionals, and vocational training students in the healthcare field.

The elected speakers, as collaborator of JJNN, were already experienced in teaching activities (seminars and other talks), or they were professionals working on the field they explain in their presentation. So, they did not have many problems in the preparation of the performance, and they could answer all the questions that the audience had.

Once the presentations were finished and before closing the course, a wide turn of questions was opened during which the attendees jointly sent their queries on a subject of such relevance.

The attendees provided positive feedback taking home the two main messages of the course about i) understanding the basics of nuclear technology can reassurance the patients before a medical test ii) the advantages of Medical Applications of Nuclear Technology greatly outweigh their disadvantages as they can help healthcare professionals saving lives.

Figure 6. First edition of the course in 2018.

Acknowledgment

Jóvenes Nucleares thanks all the people involved in this project for their participation and altruistic dedication to it, as well as Rafael Miró, Belén Juste and Alegría Montoro for their work reviewing the technical content of this course making possible its development.

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How the Nuclear industry of highly economically developed countries must prepare for the shifting paradigm in attitude of the next generation workforce

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I. Introduction

Gen Z. Commencing in around 1996, signaling the demise of the Millennials, and ending in approximately 2015, to usher in Generation Alpha, their age makes Gen Z the graduates and the apprentices currently arriving into our industry. Also referred to as Generation Snapchat, Generation Snowflake or the Digital Natives, there is no question that the workforce of the future will be vastly different to any that have come before them. But just who are these up and coming leaders of our industry, and how will that industry adapt to meet their needs?

This paper will consider just who Gen Z are, how they differ from those that came previously and what influences them. It will then explore their expectations and how to attract this generation to the workforce and the nuclear industry and finally will investigate the future of the industry and the outlook and knowledge management that will enable Generation Alpha and beyond to continue their work.

II. Problem Definition

Each new generation enters the workplace with a different world view to their predecessors. The problem of the nuclear industry lies in understanding how it must adapt the culture to this generation's new perspective and leverage their skills to its advantage. However, the nuclear industry is unique in the timescales it must consider and the data, processes and knowledge which needs to be retained throughout that time. Therefore, unlike other industries, adaptations need to consider not only the current generation, but those who will follow. Further, there is a much greater requirement for continuity of skills, so unlike many tech industries, nuclear cannot cater exclusively to the needs of the young and must allow for knowledge management and transfer from the existing working generation.

III. Who are Gen Z and what makes them different?

Generations come and go roughly once every 20 years. Often defined by a major event in history, such as the post war 'baby boomers', but sometimes simply by an arbitrarily defined period of time, generations have become a means of characterising parts of the human populace by their demographic range. Generation Z are the latest defined generation. The eldest members of Generation Z were around 23 years old in 2019 [9]. They are the recent university graduates or apprenticeship completers currently working within our industry, but the youngest of this generation, are currently only aged 3. This means that Gen Z account for a twenty-year working window, and approximately half the work force and the industry leaders for two decades.

They are characterised in a number of different ways. Most notably as the generation of technology. Gen Z didn't need to adapt to technology like their predecessors. They were born into it and used it every day. Gen Z never experienced dial-up internet, they never owned the Nokia 3310, and a 2018 YouGov study found that only one third of British schoolchildren aged between 6 and 18 could correctly identify a floppy disc (with several of the incorrect answers identifying the object as 'a save icon') [10]. The eldest of these children will be September's University intake (or the first years just arrived in Australian Universities last month). Some of them may already work for us.

But what they are familiar with, is high speed internet, smartphones, the cloud. They have grown up in a world which is a web of connectivity growing ever more complex every day. Estimates suggest that 92-95% of Gen Z have a digital footprint. In 2017, the global populace took 1.2 trillion photos, that's 160 photos each for every single person on the planet [2]. There are an estimated 5 billion mobile devices active in the world, of which a little over half have smartphone capability [7]. In developed nations, Gen Z have never lived in a world without instant access to knowledge, data and information.

What this means, is that Gen Z within these highly developed nations have a different outlook. An outlook borne out of the technology that Generation X (1965-1979) and Millennials invented and popularised. They see the world as a much more globalised entity, simply through the access they have to it. The world is a smaller place to them, but what this access to technology and information enables Gen Z to see, more clearly than anyone could have seen it before them, is the infinite possibilities. The connectivity, the rate of development, the innovation occurring across the globe is at their fingertips. And this global access creates another more subtle difference between Gen Z and their predecessors.

Diversity, inclusion, culture. It's not a coincidence that these have all become buzzwords as Gen Z move into the workplace. They have grown up with easy access to other cultures. The likelihood is that they have more experience of human diversity and therefore have a pre-built acceptance of others and their differences. Gen Z are also much more vocal about the expectations they have of their employers to share these values [6]. Many organisations now have networks or groups arising to promote inclusion, whether these be networks for race, gender, sexuality, physical and mental health or any other feature which distinguishes groups of humanity. And that's probably a good thing, because the workforce now entering into the workplace are those who'll shout the loudest for the level playing field they are trying to create.

The final thing to note about Generation Z, is their struggle with no struggles. Statistically, the world has never been 'better' than it is today. We have lowered crime rates, exponentially developed technologies and cured diseases, and yet rates of depression in the developed world are at one of the highest levels of the past century. It suggests that Gen Z are suffering a crisis of purpose. They are trying to forge an identity and need to identify the challenges they want to face and the changes they want to be remembered for. They need to find their hope and aspirations for the better world they will leave behind, just like their parents and grandparents did for them. Here may lie an opportunity for the nuclear industry.

What with their 'phone addictions', 'unrealistic expectations' and ability to be 'offended by everything', it's easy for their predecessors to look upon Gen Z with cynicism, anger or even despair, but there's no getting around the fact, they are going to run the nuclear industry and the world. Their outlook is different, but maybe they don't belong to Generation Disappointment, maybe they belong to Generation Hope.

IV. How do Generation Z feel about the nuclear industry?

The opinion of the nuclear industry held by Generation Z, will be influenced heavily by the media coverage they see. Clear memories of 9/11 make terrorism a priority issue for Millennials. However, Gen Z, the oldest of whom where barely more than toddlers when the attacks occurred, have been influenced by the major stories of their own upbringing. Unfortunately, much of the media coverage of nuclear is not positive and since people often assimilate information which may ultimately be unrelated, the extensive and long running coverage of nuclear proliferation may colour opinions of the entire industry. Forbes lists nuclear proliferation as one of the top 11 environmental concerns for Generation Z [5].

Further concerns can be instigated by the way media cover a nuclear incident, elaborating on concerns, dangers and the damage potential, rather than the actuality of the situation. The tsunami which destroyed the reactors at Fukushima still makes headlines 8 years later, discussing the radioactive release into the ocean and the removal of top-soil. Much less attention was given to the immediate death toll resulting from the radioactive release, 0.

Therefore, if news outlets are your primary or only source of information surrounding the nuclear industry, your opinion is being swayed by almost exclusively negative views. Therefore, Generation Z, the overwhelming majority of whom are exposed to vast quantities of news and information, but not at all to the nuclear industry, are liable to form a negative opinion. Those who pursue science in academia may find this opinion being changed, but one of the challenges faced by the nuclear industry is encouraging the younger generation to perceive the industry in a more positive light and to advertise the myriad of successes and opportunities it offers. It is to adapt into appreciating and leveraging their technical proficiency and demonstrate that the ethics and values which they prioritise are shared by the industry they would be working for.

V. How does the Nuclear Industry prepare for tomorrow's workforce?

So how does the industry of today prepare for, and attract, the workers of tomorrow? The first thing is to understand the objectives of Gen Z as they enter the workplace. Millennials traditionally sought 'softer' things out of their working environment, flexible hours and a gentler culture. But Millennials were, for the most part, raised during an economic boom and did not seek money as their primary priority. However, Gen Z, a product of global financial crises and recession, have more practical goals and will scrutinise salaries much more closely [3]. They care less about workplace perks, and much more about a living wage, recognising potential economic scarcity. That does not mean that there is no place for making the workplace attractive to potential recruits. With Google, Apple, Facebook, Tesla and the Walt Disney company all ranking within the top 10 most sought after companies to land jobs on LinkedIn, creating an aspirational and exciting brand is clearly still valuable [1].

To an even greater degree, a workplace must offer a purpose. Generation Z, are searching for the 'mark' they want to make on the world. They are hunting for a legacy. Every generation does. We develop eras of industry, advancement, those characterized by great struggles or social change. The UK industry adoption of steam locomotives in the mid 1800's, the two world wars which affected the lives of millions across the world, the 1960's-1980's which drove several liberating social changes across the US, from African American rights to the sexual revolution. Generation Z are searching for their own revolution. In recent years we have seen the growing popularity of pride, to celebrate a growing acceptance of LGBTQ+ rights and extinction rebellion trying to drive environmental change. But there is nothing to say that the safe advancement of the nuclear industry could not be the revolution this generation are searching for. The safe disposal of nuclear waste is a critical problem which we cannot choose to ignore. In order to attract the next generation, the nuclear industry must demonstrate this problem to them and show them how it could become their purpose.

The culture of a workplace must also adapt to the specifications of Gen Z. Their requirement for inclusivity extends to a desire for absolute workplace equality. The era of privilege, which, whilst by no means common in all workplaces, has been characteristic of some, will not survive this more community minded generation. Indeed, 77% of Generation Z state that the diversity of a workplace would have a significant impact on their decision to work there [3].

Technical advancement, in both recruitment and within the workplace is similarly critical to how Gen Z will view their potential employers. They want to see innovation, all of the potential that the world they grew up in promised, realised. They have grown up, phone in hand and experience the seemingly limitless power of a handset that can be purchased on the high street. The nuclear industry presents one of the most complex challenges in the world. The innovative technologies we offer should be absolutely extraordinary. We know the technology exists, the challenge is integrating it into our businesses. But more than this Gen Z expect creativity in the way this technology is used. Agile working will become the norm, especially in the nuclear industry, where, despite the mission remaining constant, the immediate challenges we face each day change. The use of technology will adapt with these changes and Gen Z, with their existing technological fluency, are likely to be the workers who find these adaptations.

When interacting with the younger half of Gen Z, those still at school, we may already have some capacity to demonstrate these technological innovations to them. STEM programmes operated as part of regional youth projects and internal networks are a golden opportunity to showcase our uses of technology and demonstrate how exciting and varied work in the nuclear industry can be. The nuclear arm of Jacobs Engineering is currently developing a program aimed at 15-16 year old students, in which they develop and race a state-of-the-art simulator on a circuit. The 3-4 hour session requires many of the engineering, project management and strategic thinking skills which would be required in the workplace, but in the context of something relatable and exciting.

But Gen Z will also be influenced by technology and especially by social media. In a 2019 study, 54% of Gen Z listed social media as their main brand related influence [11]. Since a huge part of recruitment is making your business brand recognisable and attractive to potential recruits, a social media presence is becoming less and less optional as a recruitment tool.

VI. Collaboration, Knowledge Management and Playing the Infinite Game

This final section will explore how the nuclear industry differs from others, but how it can still move forwards. So often, an industry is competitive. Microsoft and Apple compete for business. The Walt Disney company has competitors too, the movie studios; MGM and Paramount, the theme park owners; Universal and Six Flags and dozens of others. They are all intent upon keeping their business afloat, with little consideration for any of the others. This is where the nuclear industry differs in a fundamental way. Every facet of the nuclear industry needs to collaborate.

Nuclear power generation unquestionably has the potential to be a very dangerous game. Three Mile Island, Chernobyl and Fukushima demonstrate the significant consequences that can arise from an incident, but beyond the danger posed by these is an even greater challenge for attracting Gen Z, the perception of danger. We work in a largely, inherently misunderstood industry. Nuclear is unpopular in many factions of society. Perceived as dangerous and not worth the risk, potential consequences are often mis-judged as catastrophic. The incident at Fukushima is responsible for kickstarting the nuclear shutdown currently occurring across Germany. There is no denying that the next major nuclear incident could be a disaster of unparalleled proportion, but the risk of an incident is routinely ramped-up by the media to which Gen Z have access. As this paper is being written, a news headline on a phone notifies of an unsuccessful cyber-attack on Kudankulam, India's largest nuclear power plant. Tens of thousands of these attacks occur on industries and businesses every day, but they are not newsworthy. It is the perception of danger within the nuclear industry which makes the headline.

The nuclear industry must therefore do all it can to prevent an incident from occurring, because a single incident or failure anywhere in the world impacts everyone. 'If one operator sneezes, everyone catches a cold'. When incidents make national headlines, we are potentially driving away the next generation. They are very invested in the values of an organisation or industry and whether or not those values align with their own. Further, Gen Z are more selective about where they work than Millennials, who are traditionally more flexible about accepting employment. [4].

The other major adaptation which the nuclear industry needs to make, is playing the infinite game. Business is too often treated like a finite game. We know the rules, we know the other players, we know the end objective and that objective is to meet the end of this quarter with the profitability and safety performance expected by society. In effect, we'll deal with next quarter when we get to it. But the reality of the nuclear industry is that no matter how businesses want to treat it, we are all playing an infinite game that we need to recognise. In the infinite game, we do not necessarily know all the players and no-one has agreed the rules [8]. In the nuclear industry, we don't know how foreign policy might affect who remains involved, nor do we know the technical challenges we might face. What we do know, is that we don't 'win' for the next several hundred thousand years. No nuclear incident this quarter, does not prevent one the next and consequently we must constantly look to the future.

The technological fluency and past experience of industry upon which Gen Z will call should assist enormously with this task. Learning over years has taught us the importance of knowledge management on a site. In the past, construction occurred for the convenience of the present. Now that present has become the past and these facilities are in the decommissioning phase, it is recognised that little to no thought was given to how this might be achieved. Armed with this knowledge, the processes associated with nuclear construction have changed. When a building is constructed, we consider knowledge management and the information which will be required to decommission at the end of its life. This will not be a set of instructions, because, as Gen Z already recognise, technology is moving at an accelerating pace. Instead the tangible history of the building; it's construction materials/process, chemical signatures of the contents, risks and hazards etc, will be logged to allow future generations to leverage their own technology to make these decisions.

In conclusion, the nuclear industry, just like every other workplace, needs to recognise the changing demands of the world and its workforce and adapt to meet them. Whilst the mission of safe power generation and waste disposal never changes, if the nuclear industry expects to acquire and retain the talent of Gen Z, it must recognise their unique skills; their technical fluency, creativity and attitude of equality. Gen Z are different to the generations which came before them, much like every other generation has been different too, but in the information age, they may be the first with a very real chance of playing the infinite game.

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Communicating With The Far Future

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I. Introduction

In 1991 a group of scientists, anthropologists, architects and science-fiction writers gathered in the New Mexico desert at the request of the United States Department of Energy (DOE) to answer a single question: how best to protect buried radioactive waste from human interference for 10,000 years?

II. The Problem With The Solution

While a Geological Disposal Facility is designed to isolate radioactive waste from natural hazards, these barriers would not withstand active human interference such as drilling. A concerted effort to drill or mine for minerals around the GDF by a future society could impact the integrity of the physical protection barriers [1]. It is even possible in this hypothetical future society that some information has been passed down regarding the caverns filled with mysterious treasure that the people who came before tried to hide. After all, if it isn't valuable why would past civilisations have tried to hide it? Therefore, should the GDF be protected by a warning system?

It was this issue that the US government tasked the eclectic New Mexico group to solve in 1991. They were contacted as part of a study by Sandia National Laboratories (SNL), a US DOE contractor, to design a system of warning markers that could communicate the hazard of radioactive waste in a form that a future society could understand.

III. The Problem With The Solution

SNL designated two panels of experts: the 'Markers Panel' and the 'Futures Panel'. The Futures Panel was to investigate and predict the possible paths that society might take in the next 10,000 years and the Markers Panel was to design a warning marker system that was capable of conveying information to any future society predicted by the Futures Panel.

Ten thousand years was chosen as the required time period as this was the regulatory requirement and it was considered that the 100,000-year requirement was too onerous for an initial study into the marker's effectiveness.

The Markers Panel was split into four teams to ensure a range of options would be generated and highlight areas where the teams arrived at the same design or disagreed on the effectiveness of other designs. These comparisons would form the basis of further investigative work. The remit given to both teams was as follows:

1. The time frame for the panel to consider must be 10,000 years because of the requirement that performance assessments cover a period of 10,000 years after closure of the disposal facility.
2. The markers must be developed with a goal of being able to convey information to any future society (considering the broad spectrum of possible future societies developed by the Futures Panel [2]).
3. To communicate the dangers associated with the waste buried at the WIPP.

The teams presented their findings to SNL in the 1992 report "Expert judgement on inadvertent human intrusion into waste isolation pilot plant" [3]. All the teams assumed that there is potential for much change over the next 100,000 years and it is possible that knowledge of the GDF and what it contains may be lost. The languages spoken are also likely to change significantly so the messages cannot be written only in English, or any other language currently in use. In order to convey the content of the warning, the message must be designed to communicate at a level beyond written alphabetical language.

If the message is to remain intact during the lifetime of the GDF it must be comprised of erosion resistant materials or located underground to preserve it. The material should not be considered a valuable or useful resource in case it is looted or repurposed for building material. The message must be capable of conveying three parts:

1. That there is a message at all.
2. That hazardous substances are located in this area.
3. Information about the hazard.

IV. Communicating With The Far Future

Symbolic messages are capable of carrying a lot of information but do not resemble the signifier that is being represented. They are learnt culturally and rely heavily on context (Figure 1). Symbols are widespread in our culture from the Golden Arches to the hammer and sickle to the skull and crossbones. Each of these symbols convey an array of meanings depending on the reader and where it is seen.

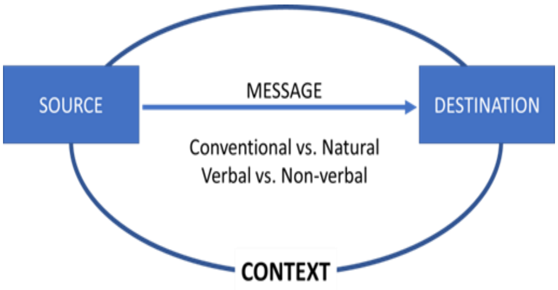

Figure 1. Types of messages in relation to context.

However, these are relatively recent meanings in comparison to the expected lifetime of the GDF. The skull and crossbones (Figure 2) in particular has had a variety of meanings from its origins in medieval paintings, to piracy, to denoting poisonous substances. It is very likely that well known symbols like these will continue to evolve over the centuries. Therefore, they cannot be relied upon to accurately convey information to future society.

Figure 2. The skull and crossbones/Jolly Roger. An example of a symbol with changing meaning..

V. A Marker System For The UK

What does all this mean for the UK's plans to build a geological disposal facility? Unlike the US there is no UK regulatory requirement to construct a warning marker system over a GDF that will last longer than the operational life of the facility. The IAEA specific safety requirements for disposal of radioactive waste include a requirement to reduce the long-term risk of intrusion after the closure of the facility "by the use of passive controls, such as the preservation of information by the use of markers and archives, including international archives" [3]. The IAEA also foresees that the responsibility for passive measures for institutional control will revert to the government after the site is de-licensed.

As seen in the UK government's decision to request volunteer communities to host the GDF, any arrangement must be beneficial to the hosting community in some way. A cost benefit type analysis for any future generation marker solution is complex and this may also consider the immediate and future benefit to the community. As shown by the SNL study, estimating the structure of future society is incredibly difficult yet vital for the determination of the correct level of effort and finance to invest into design and construction of a marker system.

This poses the question of whether the best option is to obscure the location of the waste facility. Local stakeholders could be compensated in return for moving from the area in a conscious effort to prevent memorisation of the area as location for nuclear waste. Temporary signage and barriers could be designed to erode and rust over the course of a few centuries after active control of the site stops.

However, assuming the worst-case scenario projected by the Sandia team (a society with 1800's level exploratory drilling technology with little knowledge of radiation and radioactive waste) the chances of accidental discovery and intrusion are non-trivial. In the US (and Russia, China and Australia) such a "fill and forget" approach may work technically, if not morally or ethically. The UK does not have a large enough landmass to effectively hide the facility over the timescales required. Currently the UK has a population density of 710 people per square mile, by comparison the US and Australia have 87 and 9 people per square mile respectively. The Sandia teams assumed that any event which would cause societal regression would either not impact population density too severely or that it would recover to close to modern levels by the time modern drilling equipment is re-developed.

If a marker system was judged to be required, the Government should encourage collaboration with industry and academia to design a minimal but functional marker. This "artefact" must have an immediate and useful purpose. The obvious suggestion would be for it to contain educational messages and materials that will allow visitors to learn about radioactivity and nuclear waste. This would effectively demystify radiation as well as provide a local source of tourism income. If the design is sourced locally it will give the community a sense of ownership over the artefact and improve relations with the GDF operating company. Ideally, the funding for construction should come from a mix of private and public sources so there is not the appearance of 'wasting' money that would be better spend on local infrastructure.

VI. Conclusion

Perhaps then, the solution is as simple as building a visitor centre. A well-equipped and well-staffed visitor centre would fulfil many of the above requirements and could be incorporated into a meaningful and long-lasting structure. The centre would provide long-term education and employment to the local community and be a visible symbol of the benefits of hosting a GDF. The structure would endure after the closure of the GDF and visitor centre to act as a warning marker containing information about the radioactive waste buried deep below. While there is still active control over the site the structure can be maintained and expanded providing further employment for the host community and improving the probability that all or part of the structure will endure over the required span of thousands of years.

What is required now is further multinational development of the design and analysis of the most cost-effective warning system taking into account both current society and future generations. The UK government and nuclear industry has both the experience and expertise to launch and lead such a venture in conjunction with other GDF countries and organisations such as the NEA and IAEA.

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State of the UK Nuclear Industry

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I. INTRODUCTION

The UK's involvement in the science of nuclear processes can be traced back to 1919 when New-Zealand born Ernest Rutherford, now known as the "father of nuclear physics", bombarded a nitrogen target with alpha particles from a source at the University of Manchester. He observed that the target produced protons (which at the time he called Hydrogen nuclei) and transmuted the nitrogen to an isotope of oxygen [1].
One of Rutherford's students James Chadwick identified a sub-atomic particle in 1932 with no charge which he called the neutron.

Since then the UK has been deeply involved in the nuclear industry in all it's forms and aspects. In 1940 the UK hosted the MAUD Committee which calculated the critical mass of uranium required to make an atomic fission device was less than 10kg rather than the 40tonnes (12tonnes if using a neutron reflector) that had previously been assumed. The report put atomic bombs within the realms of possibility and paved the way for the development of nuclear weapons. In collaboration with the Canadian and American governments the British government began work on developing an atomic bomb. The project was code-named "Tube Alloys", however after lack of progress this project was eventually shelved, and the scientists and materials moved to the US to contribute to the "Manhattan Project".

The MAUD committee also released a second report "Use of Uranium as a source of power" which discussed how useful energy might be extracted from the nuclear process to provide power and possible designs for a nuclear power station.

In the aftermath of World War 2 the British government then examined whether nuclear reactors could be used to produce plutonium for its own weapons programme. A side effect of these reactors was that the heat produced could be extracted and used to boil water into steam which could then move a turbine to generate electricity. In 1956 Calder Hall Operating Station became the first commercial nuclear power station to be connected to the grid.

Using data and information gathered from the initial test reactors the UK built a fleet of graphite moderated, gas cooled reactors called Magnox after the Magnesium-Non-Oxidising cladding used in the fuel elements. They were designed to produce cheap electricity for British homes and industry at the same time as producing plutonium for the ongoing weapons programme. The government intended for the design to be exported and built overseas but by the 1970s design flaws, construction delays and competition from LWRs meant that only two Magnox reactors were built overseas.

The requirement to produce plutonium had limited the efficiency of the Magnox reactors. The next Gen II reactor designs kept the graphite moderator and CO2 coolant but increased the coolant temperature to create more efficient steam conditions. The Advanced Gas-cooled Reactors became the backbone of the UK nuclear industry with 14 built in between 1976 and 1988 [2].

II. THE "NUCLEAR RENAISSANCE"

From 2028 the UK's existing fleet of AGRs will be taken offline and by 2035 only the Sizewell B PWR will still be operational. Therefore, a programme of nuclear new build is needed to replace these ageing reactors, ideally before they are shut down for the last time [3].

At the moment the UK's only civil nuclear new build under construction is Hinkley Point C, currently in the middle pouring nuclear concrete. The project relies heavily on overseas workers particularly from Europe, India and China. The Chinese company China General Nuclear (CGN) purchased a 33% stake in the HPC project and will support the next EPR new build at Sizewell before developing their own design, the Hualong Power Reactor, at Bradwell. Currently the HPR is undergoing the Generic Design Assessment by the UK regulators ONR and EA with a final decision expected in 2022.

As part of the EU the UK has also contributed towards the iter project in Cadarache, France and this collaboration is not expected to end after the UK leaves the EU.

The UK government has also announced funding for research on fusion and Small Modular Reactors (SMRs). The Spherical Tokamak for Energy Production (STEP) programme aims to build on the UK's experience on the Joint European Torus (JET), iter and MAST fusion projects to develop a national fusion reactor capable of supplying 100MW of electricity by 2040. STEP will use £20m to develop an integrated concept design

As in many countries there has been a surge in interest in building Small Modular Reactors in the UK. The government announced £56m for funding into SMR research for eight companies and in 2019 awarded the UK SMR consortium £18m to develop the 440MWe design. The consortium is made up of a number of well-known nuclear companies including Wood, Rolls-Royce, NAMRC and the National Nuclear Laboratory. The consortium plans to move the design from first concept to ready for regulator assessment over the next 24 months.

III. DECOMMISSIONING

In addition to the need for new nuclear power stations the UK must also develop and increase its decommissioning capabilities. As the Magnox reactors, and eventually the AGRs, have gone offline there is an increasing need to devote more resources to the ensuring that they are decommissioned in a safe manner compliant with regulations. The Nuclear Decommissioning Authority (NDA) is responsible for managing the decommissioning of Magnox reactors and facilities at Sellafield. The cost of decommissioning such a wide range facilities over 19 sites has been estimated as high as £73bn [4].

Decommissioning does however present an opportunity for the British government. Due to Britain's early and extensive involvement in the nuclear industry it has a wide range of different sites with unique challenges that are approaching decommissioning. If managed properly this could allow Britain to develop a centre for excellence for decommissioning which could then export its expertise worldwide. Many other countries who developed nuclear power later than Britain will not have to decommission their facilities yet allowing the UK to practice and perfect decommissioning processes and techniques before offering its services.

After the existing reactors have been defueled there remains the issue of radioactive waste disposal. Previous attempts to build a Geological Disposal Facility in North West England have failed as the local county council vetoed the proposed facility in West Cumbria. Currently Intermediate Level Waste (ILW) and High Level Waste (HLW) is stored at Sellafield on an interim basis while a decision for final disposal is debated. A new site is expected to be selected by 2040 with interested local communities volunteering to host a site. These options will then be investigated to ensure the proposed site geology is satisfactory for constructing the GDF. The intention is that this facility will be able to store waste produced from new build sites as well as the legacy waste from over 60 years of power production.

IV. PUBLIC OPINION

Overall UK public opinion has been reasonably favourable to nuclear power over the decades. A majority of the public are in favour of continued use of nuclear power although the proportions are heavily affected by gender, age and socio-economic background. The most typical supporter of nuclear power is males aged 65 or older who earn over £50,000. The main reasons for support of nuclear power are "affordability" and "reliability" with only 37% agreeing that it has a role to play in combating climate change [5]. Encouragingly support for nuclear is high amongst communities in close proximity to nuclear power plants and proposed new build sites.

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Professional public accreditation of educational programs implemented by ROSATOM

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I. INTRODUCTION

In the modern world, the level of technology development is constantly growing. The education system takes some time to adapt to new conditions. Educational programs require constant optimization and updating, in accordance with the requirements of the labor market.

In order to improve the quality of training of specialists produced by educational institutions, the Russian Federation implements a system of professional and public accreditation of educational programs.

Accreditation of educational programs of educational institutions is a form of feedback for the management of the educational institution. This procedure allows you to analyze the results of decisions made on the organization of the educational process and its effectiveness, which is extremely important for achieving the goals.

II. PROFESSIONAL PUBLIC ACCREDITATION

Analysis of the actual situation, improving the quality of education and increasing its practical orientation are priority areas of activity of the entire educational structure. These activities have an impact on the level of training of specialists produced by higher education institutions, their effectiveness in participating in work processes, as well as their competitiveness in the labor market.

In turn, the results of the activities of specialists issued by educational institutions form the current state of the relevant industries and the economic situation throughout the country.

Professional-public accreditation of the basic professional educational programmes of basic vocational training and (or) additional professional programs is a recognition of the quality and level of training of graduates who have mastered the educational program of each organization performing educational activities that meet the requirements of professional standards, labor market requirements specialists, workers and employees of the corresponding profile [1].

Professional public accreditation includes several stages:

1. Formation of a group of experts (a specific area of knowledge);

2. Development, discussion and approval of educational program evaluation criteria;

3. Collecting applications from higher education institutions;

4. Formation of a working group by selecting several highly specialized experts from the General group, depending on the specifics of the analyzed discipline;

5. Self-examination of the educational program of the educational institution according to the accepted evaluation criteria;

6. Setting deadlines for the professional and public accreditation procedure for a specific curriculum;

7. Formation and demonstration of a self-examination report by a education institution;

8. Familiarization with the report on self-examination of the Commission, analysis of the report;

9. Making recommendations by experts to improve the program;

10. Departure of experts to the higher education institution, evaluation of the training program (familiarization with documents, material and technical resources, software, communication with the administration, students, teachers, technical specialists who serve the educational institution;

11. Meeting of experts, putting down points according to the evaluation criteria;

12. Familiarization of the management and employees of the higher education institution with the results of the inspection;

13. Development of recommendations by experts to improve the actual state of the curriculum;

14. The adoption of a positive or negative decision about the accreditation of this educational program;

15. Issuance (in case of a positive decision) of a certificate of accreditation.

ROSATOM is a large state Corporation that includes various divisions that specialize in certain stages of production processes. ROSATOM has a developed administrative and management system. ROSATOM cooperates widely with educational institutions of the Russian Federation. Educational programs for employees are also implemented within the Corporation itself. The range of opportunities for professional public accreditation of educational programs of educational institutions of the Russian Federation is quite diverse.

III. CONCLUSIONS

Conducting professional and public accreditation procedures that analyze the level of proficiency and the direction of development of competencies received by students of educational institutions allows reducing the financial costs of retraining and retraining of specialists produced by educational institutions, which arise due to the natural obsolescence of educational programs, due to the intensity of technology development.

Since ROSATOM uses the latest technologies, software, and experienced specialists, conducting professional public accreditation to optimize educational programs can be of great benefit to improving the professional level of specialists produced by educational institutions. The accreditation process allows you to provide up - to - date feedback from employers-consumers of educational products to educational institutions-suppliers of educational products.

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TRACK 10: NON-POWER APPLICATIONS: MEDICINE, BIOLOGY AND INDUSTRY

PROJECT FOR THE DEVELOPMENT OF TELEMEDICINE TECHNOLOGIES IN THE ATOMIC CITIES AND CREATION OF THE TELEMEDICINE CENTERS AT THE CITY CLINICS («MEDIO»)

KRISTINA IZBITSKAYA
ROSATOM, RUSSIA

BRINGING NUCLEAR TECHNOLOGY TO SPACE

A. PAQUET, F. NOUCHY
TRACTEBEL ENGIE, BELGIUM

INVESTIGATION OF DNA DAMAGE VIA GAMMA-H2AX ASSAY COMBINED WITH MICRODOSIMETRIC ANALYSIS

UI-SEOB LEE AND EUN-HEE KIM
SEOUL NATIONAL UNIVERSITY, SOUTH KOREA

TOXIC METAL MANAGEMENT WITH FUNCTIONALISED TITANIUM DIOXIDE

L. HALLAM AND J. VELISCEK CAROLAN
ANSTO, AUSTRALIA

ENHANCING MATERIAL PROPERTIES AND RADIOTRACERS BY DEUTERATION

R.B. MURPHY, N.A. WYATT, B.H. FRASER, N.R. YEPURI, P. J. HOLDEN, A.T.L. WOTHERSPOON, AND T.A. DARWISH
ANSTO, AUSTRALIA

A SIMULATION STUDY ON DETECTING FAKE GOLD BAR USING NEUTRON IRRADIATION-BASED ANALYSIS

KI-MAN LEE AND GWANG-MIN
SUN KAERI, SOUTH KOREA

FROM SKETCHES TO SALES: LESSONS FROM THE DELIVERY OF A NUCLEAR MEDICINE PROCESSING FACILITY

RYAN BEEBY
ANSTO, AUSTRALIA

PHARMACEUTICAL RISK MANAGEMENT FOR RADIOPHARMACEUTICAL PRODUCT AND PROCESS DEVELOPMENT

KAITLYN GUNDERSON-BRIGGS

ANSTO, AUSTRALIA

MAINTENANCE FACTORS AFFECTING INCREASED LUTETIUM-177 PRODUCTION

NIKLAS DANIELSSON
ANSTO, AUSTRALIA

INTRODUCTION OF AN ISOTOPE PRODUCTION SYSTEM TO CANDU REACTORS

S. DOS SANTOS, R. SIVAKUMARAN, K. BROWN, W. COOPER
KINECTRICS INC, CANADA

DEVELOPING AND MAINTAINING INTERNATIONALLY TRACEABLE RADIONUCLIDE STANDARDS TO BENEFIT AUSTRALIAN NUCLEAR MEDICINE AND INDUSTRY

S. LEE, F. VAN WYNGAARDT, M. SMITH, T. JACKSON, B. CARUANA AND C. KEEVERS
ANSTO, AUSTRALIA

STUDY OF ENVIRONMENTAL SAMPLES UTILIZING THE NEUTRON ACTIVATION ANALYSIS AT THE TRAINING REACTOR VR-1

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RADIOSENSITIVITY OF *ALOCASIA MAQUILINGENSIS* MERR. SEEDS

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SYMO: AN AUSTRALIAN FIRST OF A KIND WASTE TREATMENT PLANT UTILIZING SYNROC TECHNOLGY

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Project for the development of telemedicine technologies in the atomic cities and creation of the telemedicine centers at the city clinics («MEDIO»)

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I. Introduction

Rosatom is implementing a variety of educational programs, development programs and mentoring, including programs for young employees of the industry. Probably, this is one of the reasons why Rosatom was recognized as the best employer in Russia for 2018. In August 2018, the author became a member of the first Rosatom Youth Congress where work was organized in seven designated areas. These areas are the key to the development of the industry. In the track “Development of Atomic Cities”, where among 11 projects developed and presented at the Congress, our team, which initially consisted of two people, managed to reach the finals of the contest and presented our idea to the Director General of Rosatom.

II. Description of the Task

This paper describes the content of the project and the results achieved. The aim of the project was to identify key factors for the development of so-called "Atomic Cities" based on an analysis of survey data and to develop a project. Before coming up with the idea, the team conducted a small study among the participants of the Congress and asked to identify the growth points of our cities, which in their opinion are the most significant and important. These results are shown in the following figure.

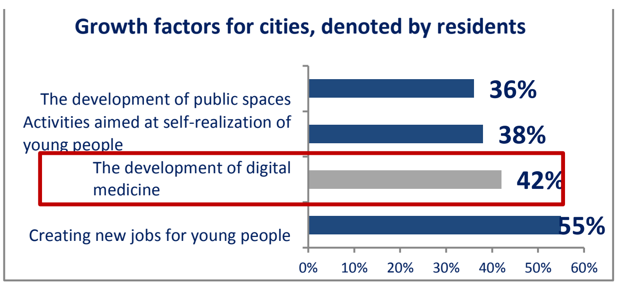

Figure 1: Growth factors for cities, denoted by residents

A. Atomic Cities

Atomic cities cities are cities where Rosatom is present, that is, cities in which nuclear power plants are located as well as uranium enrichment plants, etc.

These cities with a small population, and these cities are located far from large administrative centers. These cities often lack doctors with narrow specializations, and it takes time for the citizens to get medical help and advice. Doctors from bigger cities rarely move to nuclear cities, even though Rosatom and the city administration create favorable conditions for the doctors, who decided to move to these territories. For instance, housing is being provided for young medical specialists. Funds are also provided for a favorable start in life. In short, various compensations are offered, but only 1 doctor out of 10 decides to move to the atomic city for work. And often as practice shows, only for a short time. [1]

B. Formulating the task and methodology

The team studied what can be offered to improve the situation in nuclear cities in the field of medical care and turned the attention to the development of telemedicine technologies. Telemedicine in Russia is at the beginning stage of the development and was only adopted in the Federal Law early 2018. It recognized telemedicine as the legal form of medical services. Having studied the material, the team proposed to implement a project to create telemedicine centers at polyclinics in the cities of Rosatom. The first, small step was positively perceived by the CEO (Director General) of Rosatom and started project development. In the following figure some key points of what a telemedicine service should provide is shown.[2,3]

Creation of telemedicine centers (stationary/mobile) at city clinics, hospitals with the ability to use:

- Teleconsultations **by appointment** (mmi, vmi);
- **Real time** consultations (mmi, vmi);
- **Tele-education of medical staff;**
- Conducting **educational competitions medskills;**
- Remote biomonitoring systems (gadgets and mobile apps);
- **Creation of bigdata medical knowledge of our cities**

Figure 2: llustration of the key roles of telemedicine

MMI – Mandatory Medical Insurance VMI – Voluntary Medical Insurance

III. Description of the Project Process

A. Overview

A team consisting of young specialists has begun a difficult way of collecting vast amounts of information in the field of medical legislation, including studying Russian and international experience.[4,5] Medical specialists, professors and leaders of public medical organizations that are engaged in protecting the rights of patients were consulted. Also residents of the city were interviewed on their opinion on the implementation of this project. Finally, the team was able to form the tasks for ourselves that can be solved through the development of telemedicine technology. Considering the global trend of digitalization development, it is thought that a market has been found. Out of all the benefits that telemedicine can bring to a city, the following stand out:

- to increase the accessibility and quality of medicine;
- to continuously professionally develop medical personnel.

In the following figure some major tasks and possible solutions are shown in an overview:

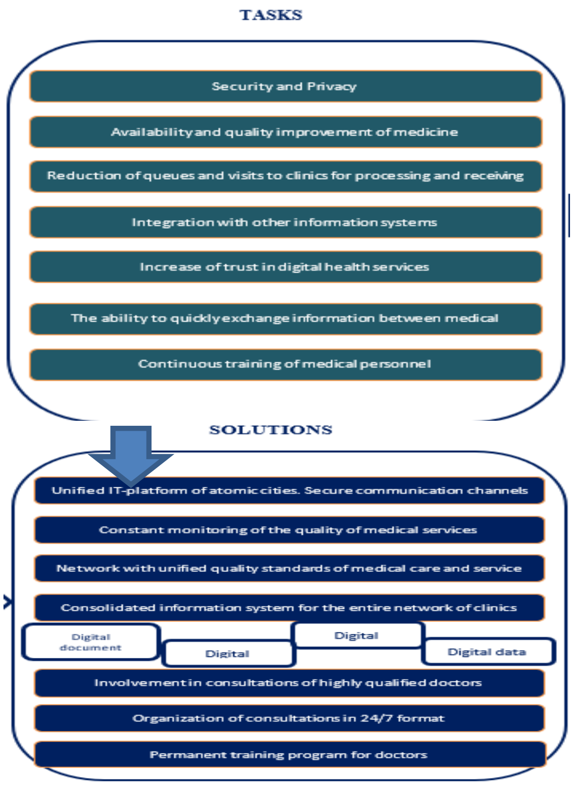

Figure 3: Overview of telemedicine principles

B. Schema

Below in figure 4 is schematically depicted what is proposed and how a similar scheme was implemented in a remote village, with the help of volunteers who participated in the contest “Leaders of Russia».[6] The particular project was recognized as one of the best at this competition. Special telemedicine equipment was purchased, which was installed in a medical office in the village. It turned out to connect physicians in the village and physicians from the polyclinic of the neighboring city. That is, in the presence of a therapist, patients can now receive advice from a wide range of specialists, without wasting time, money and nerves on moving from the village to the city. It should be noted that in the proposed project, it is planned to fully involve the doctors who are already working in the city.[7] In other cases, where this is not possible, advice from the regional and federal medical centers is sought. It should also be noted that the project has two large parts: telemedical equipment and software. It is currently being evaluated to either develop special digital solution for atomic cities or to use the solutions that already exist on the market.

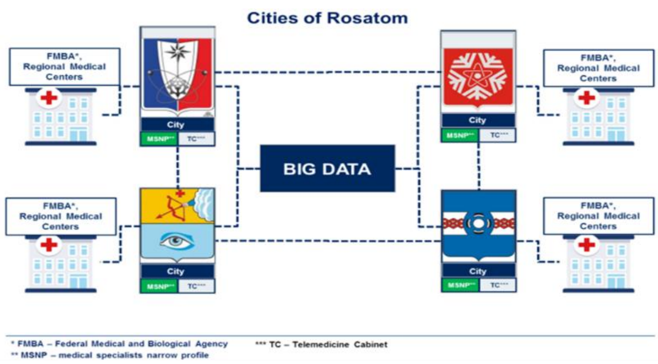

Figure 4: Scheme visualization of project

IV. Results

The project has been demanding and had to overcome many challenges and resistance. Part of the difficulty was to convince people of the benefits of telemedicine. Several events were visited since last year, where it was told again and again that telemedicine is not a story about the incredible technologies. This is a story about people and the adoption of the new approach. The benefit of innovative technology in the future cannot be neglected. If digital medicine and its developments will contribute to more lives saved, then it deserves an increase of attention to this area.

Chronology of the project

In the following figure the chronology of the project is shown.

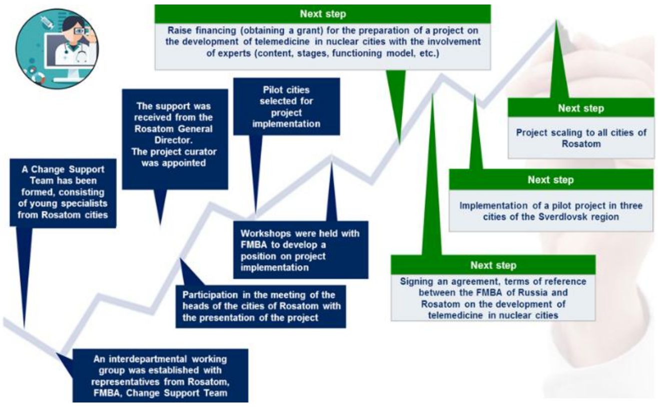

Figure 5: Decryption of the chronology of project

It was assumed that the most important thing is to manage to draw attention to the project and to the several groups of people involved in the project. The working group has grown considerably and the now includes not only representatives of the Rosatom youth, but also representatives of the main Department of Rosatom, which is responsible for interaction with the regions, the Federal Medical-Biological Agency and representatives of public medical organizations.[8]

The project is run by the youth of Rosatom, residents of Russian cities. It is important that the population of atomic cities live in an interesting and comfortable way, and elementary comfort begins with receiving high-quality, timely, modern medical care. The development of telemedicine technologies can qualitatively improve the situation. Nuclear cities are home to highly qualified personnel, world-renowned scientists and a whole milestone in history not only for Russia, but for the whole world.

The younger generation is the future of this World and it is within our power to do everything possible to solve the problems that exist now so as not to leave them to future generations. We are ready to develop cities, we are ready to live in them, we want to use every opportunity to improve the quality of life in our cities and sincerely believe that we will succeed.

Any methods of preserving life and health should be used to the maximum. Telemedicine must be developed in the regions, in the atomic and non-nuclear cities of Russia.

The next milestone in the history of the development of the project will be the development of telemedicine consultations in the "doctor-patient" format, which can be obtained free of charge as part of the compulsory medical insurance program. This will significantly increase the effectiveness of repeated doses, for example, to adjust treatment. This is especially true for seriously ill patients who are not easily given every visit to the doctor.[9]

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Bringing Nuclear Technology to Space

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I. INTRODUCTION

Nuclear engineering has always been a high-tech, stimulating, and innovative industry. A wide range of nuclear skills and competences such as radiation shielding and fluid dynamics can be applied to other technical fields. This paper focuses on the diversification of nuclear activities in space applications, where competences used in the nuclear industry can become an asset. The transition between an expertise in nuclear power generation and a skill set applicable to the space industry starts with identifying compatibilities between the two fields and assessing the gap of competences. This paper specifically addresses the first step of this diversification challenge carried on by Tractebel Engie.

II. GAP ANALYSIS

A. Overview of Competences

There is an extensive overlap between the needs of the space industry and the competences developed for nuclear activities. Nuclear science is at the root of many challenges in key areas including radiation protection, propulsion, systems engineering, thermal hydraulics, remote energy generation, materials science, etc. Tractebel is a Belgian company active for more than 40 years in the field of nuclear engineering and, until now, with no experience in the space field. The various areas of nuclear expertise at Tractebel are listed in Table I. It can be observed that many of them are also key competences in space.

TABLE I. IDENTIFICATION OF CROSS-MARKET APPLICATIONS BY COMPETENCE CENTER

Expertise	**Nuclear**	**Space**
Radiation Physics and Shielding	x	x
Neutronics and Criticality	x	
Reactor Physics	x	
Core Dynamics and Control	x	
Systems Engineering	x	x
Nuclear Power Plant (NPP) Operation	x	
Residual Heat	x	x
Fuel Behavior	x	
Materials Science	x	x
Thermal Hydraulics	x	x
Thermal Stratification	x	x
Computational Fluid Dynamics (CFD)	x	x
Hydrogen Management	x	
Analysis of Mechanical Components	x	x
Equipment Qualification	x	x
Instrumentation and Control (I&C)	x	x
Civil Engineering	x	
Fracture Mechanics	x	x
Piping Engineering	x	x
Seismic Engineering	x	x
Safety Evaluation Methodologies	x	
Severe Accidents Analysis	x	
PSA Level 2	x	

B. Methodology of the Gap Analysis

The first step in the gap assessment was to identify the needs of the space industry on which a technical gap analysis can be performed. For this purpose, a small number of tenders within the European Space Agency (ESA) were chosen and their technical requirements were evaluated using various weighted factors including their correspondence with the competence centers of Tractebel.

Among the initial selection criteria, the following points appeared of utmost importance:

- Technical readiness;
- Cross applications with in-house activities; and
- Partnerships (established and potential).

Other criteria included economic interest, multidisciplinary aspect (tenders with multidisciplinary technical content were preferred in order to involve more competence centers within

Tractebel), competition analysis, etc. It appeared clear though that for the pilot tenders, these latter aspects had to be set aside.

As one of the first obstacles was to get acquainted with the nomenclature, the projects, and the market, a preliminary tender selection based on known activities was conducted on each potential tender to assess the technical readiness level within Tractebel to compete in these tenders. The focus was therefore brought on keywords as "radiation", "shielding", "CFD" and others that could be associated to the domains in Table I. After the first selection and a thorough understanding of the subject, the gap analysis could be carried out by the experts of the concerned competence centers. For this purpose, the description of works to be performed was screened thoroughly and for each working phase the feasibility was assessed and the gaps listed.

The ESA assigns great importance to the formation of a credible consortium in its awarding procedure. In this context, it became clear that optimal partnerships are the key to winning new contracts. This philosophy of collaboration was accompanied by a clear openness of new entrants in the market. This led to performing the first steps in network building and to integrating it in the gap analysis. Indeed, this can be seen as a win-win-win situation for all new entrant companies, the ESA, and well-established companies for the increase in competition. In particular for Tractebel, working together with other companies more experienced in the space industry would help to fill the gap of competences and result in an acquisition of new technical knowledge and an increase in visibility.

A last aspect that was observed, but deliberately ignored, was that the reaction time to respond to those tenders is very short. Therefore, a high level of readiness is a necessity before the opening of a call for tender.

In the end, four tenders from the ESA were selected on which a complete gap analysis was performed to determine the ability of Tractebel to position itself on these tenders and, more generally, on the space market.

III. Results

A. *Description of the Selected Tenders*

The tenders selected with the aforementioned preliminary gap assessment are listed below:

1) The conceptual design of a space radiation reference payload for Lunar Orbital Platform (project known as Gateway);

2) The review and qualification of reliable material characterization methods for ablative materials used in thermal protection systems;

3) The design and characterization of an electric pump fed propellant supply for satellite propulsion systems, including hot fire testing; and

4) The development of physical models and algorithms, and prototyping of a multidimensional two-phase heat pipe network.

These four tenders cover various fields of expertise such as thermal hydraulics, systems engineering, mechanics, materials science, testing and qualification, and radiation protection. All these competences have been mentioned in Table I.

B. *Results of the Gap Analysis*

The completion of a full gap analysis on these selected tenders serves as an example to help identify the technical gaps between nuclear know-how and space technical requirements.

1) *Payload Conceptual Design*

The first tender focuses on shielding and detection of space radiation. The radiation field in space varies significantly from the radiation environment on Earth and is mainly characterized by high-energy protons, alpha particles and heavy, highly energetic (up to 10^{20} eV) ions. The shielding against ionizing radiation in space including solar particle events (SPE), galactic cosmic rays (GCR) and secondary radiation requires careful trade-offs between various parameters such as the payload mass and the use of low-Z materials. Standard shielding materials could instead increase the exposure by producing showers of secondary particles. Although most of the competences in radiation transport calculations for shielding optimization are acquired within Tractebel, some complex shielding issues regarding space radiation physics could require external expertise.

Detection of ionizing radiation in space applications has very specific needs to be effective. The challenges are numerous and include the monitoring functions of active dosimeters, prior to detection mode, and the power consumption allocated for operation. Since space radiation is highly energetic, continuous monitoring is necessary for radiation protection purposes, to provide exposure measures in real time. Dosimeters for space applications must be able to detect ionizing particles within lowest and highest energy ranges and should be sensitive enough to distinguish between different types of radiation. Other key design features include the ability of measuring linear energy transfer (LET) spectra and the use of tissue equivalent proportional counters (TEPC) corresponding to real tissue properties, from which the quality factors of relative biological effectiveness (RBE) can be derived. This provides an estimation of the dose equivalent as a measure of the exposure. Tractebel therefore needs additional support to tackle all these technical issues regarding active dosimetry.

2) *Material Characterization Methods*

The requirements of the second tender relate to the qualification of techniques for the characterization of ablative material properties including heat capacity, thermal diffusivity, thermal conductivity, and derivation of ablation enthalpies in the temperature range of material applications. Ablative materials are able to withstand very high heat fluxes and are not used in the nuclear industry. As a first step, a review of existing characterization methods is made to assess and verify their suitability and accuracy to reliably and repeatedly extract the desired properties on ablative materials. The goal is to identify the main technical challenges of ablative material transformations and quantify the interference effects caused by sources of measurement. Relevant improvements and adjustments are proposed to enhance the current testing methodology and a detailed procurement plan is defined. The test campaign aims to compare current and improved testing methods and identify the modifications needed for optimization of the test procedure. A verification plan is then established to consolidate the selected method.

Tractebel has a broad experience on review of methodology and standards for materials (e.g. the ASME code), but has no expertise with ablative materials and transformation phenomena. Nonetheless, it would have the potential to establish characterization methods of material thermal properties and associated test procedures as it has the expertise in materials science to analyze results and define proposals for the modification and enhancement of procedures. As an engineering company, Tractebel has also a strong experience in quality assurance (QA) follow-up, establishment of inspection and test plans (ITP) and technical requisition files (TRF), but it does not have its own testing infrastructure. Partnership with an experienced stakeholder in validation testing will therefore be needed.

3) Propellant Supply Design

The third tender is about the design of a propellant supply system fed by an electric pump. The aim is to reduce the feed system complexity, decrease the chemical propulsion system dry mass, and improve the feed pressure control, hence increasing the thrust performance. For the qualification of this new design, testing is expected to be performed under representative load conditions.

At first glance, knowledge on thruster technology is out of the scope of Tractebel activities. However, Tractebel has wide experience in the design of electric pumps and the trade-offs between technical requirements (head losses, flow rates, etc.). Market analysis, system and component (motor, valve, piping, etc.) specifications, and test procedures are all activities that Tractebel performs routinely. The only significant change in the preliminary concept definition is the chemical compatibility of the feed pump with the propellant. The detailed design of the motor (especially with regard to hardware and I&C systems) and the pump manufacturing will most likely require the help of an original equipment manufacturer (OEM). Nonetheless, both the system performance analysis and characterization of electric pumps are in the scope of Tractebel. Its broad expertise in system simulation models and equipment qualification for NPPs, along with the appropriate testing infrastructure allow proper verification of the pump performance and its compliance with specific conditions such as environmental constraints (thermal, mechanical, vacuum, etc.), pressure and flow rate requirements, and vibration generation. The last part of the design consists of the thruster breadboard setup and the establishment of a hot-fire testing plan, which are not competences developed within Tractebel. Furthermore, the final analysis and comparison of performance between different supply system designs will need external expertise and knowledge on space thrusters in order to draw the right conclusions.

4) Heat Pipe Network Modeling

The fourth and last tender focuses on the development of physical models and algorithms to simulate a multidimensional two-phase heat pipe network, the design and manufacturing of a multidimensional heat pipe network breadboard for models validation, and finally the consolidation of the validated physical models into parametric models suitable for use on future projects.

Two-phase heat pipe networks require specific knowledge that still needs to be acquired within Tractebel in order to effectively establish a complete and self-consistent set of technical requirements. Nevertheless, Tractebel has strong expertise in thermal hydraulics modeling, in particular CFD, but also with modeling heat transfer in materials. The realization of a mockup needs first computer-aided design (CAD) modeling which can be managed internally, unlike manufacturing. The validation of the developed physical models against data obtained from tests on the breadboard can also be shared among the competence centers of Tractebel. In conclusion, the main gap lies in the mockup manufacturing.

5) Gap Analysis Conclusion

For each of the four tenders, the percentage of the scope covered by Tractebel's current expertise is estimated and presented in Table II. On average, Tractebel can expect to cover approximately half of the project scope, which is a positive result considering the lack of experience in the space market. Furthermore, this percentage is expected to increase in the future as Tractebel will further develop its expertise in space applications. The experience of other Business Units of the Engie group can also play a key role.

TABLE II. ESTIMATION OF THE GAP OF COMPETENCES

Tenders	Percentage of the scope covered by Tractebel
Reference payload with space radiation protection	65 %
Review and qualification of reliable material characterization methods	35 %
Electric pump fed propellant supply for satellite propulsion systems	35 %
Multidimensional two-phase heat pipe network	55 %

IV. CONCLUSION

Space offers interesting opportunities in a market that presents attractive and long-term growth prospects. In this environment, companies that have developed broad expertise in nuclear field for a long time, such as Tractebel, have a key role to play as an engineering partner for space companies. During the market analysis made in parallel to the gap analysis, it was concluded that the hot topics in the space field include equipment qualification, thermal hydraulics calculation, and radiation shielding. These are all technical fields where a nuclear engineering company has knowledge and skills. Other space related projects, not mentioned in this paper, are too far from the competences of a nuclear engineer to be even considered, e.g. satellite communication technologies. The gaps in the feasible projects are found mostly in design knowledge and appropriate testing equipment, where establishing strong collaboration with other experienced stakeholders is the strategy to follow. This diversification plan allows a sustainable transition by progressively expanding the internal expertise to other fields of activity, gaining enough credibility and legitimacy in the space industry.

One of the other major challenges that a company faces when it comes to diversification is the lack of allocated resources. The efficiency of the organizational structure and the availability of a project team will be essential to drive results. Nevertheless, this diversification strategy in a company operating in the nuclear sector opens up new prospects and will be a major asset to attract future generations of nuclear engineers, especially those who are also interested in the space industry.

ACKNOWLEDGMENTS

First of all, we would like to express our sincere thanks to our colleagues Gert Pille and Tom Hendrix for their precious assistance and suggestions in the writing of this paper.

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Investigation of DNA Damage via Gamma-H2AX Assay Combined with Microdosimetric Analysis

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I. Introduction

Gamma-H2AX immunofluorescence detection assay is a method to find DNA double strand breakages (DSBs) [1]. Gamma-H2AX signal can be detected in a form of the fluorescent focus. By counting the number of gamma-H2AX foci, one can quantitate the DNA DSBs. The number of gamma-H2AX foci is reasonably expected to increase linearly with radiation dose. Hence, gamma-H2AX assay is utilized in measuring the number of foci in blood samples exposed to unknown doses [2]. However, the method of counting foci has some pitfalls in discerning the severity of DNA damage. Separate foci are counted the same regardless of size. Large foci, most probably attributed to the overlapping of multiple foci, are not differentiated from small foci resulting from single DSBs [3, 4]. Considering that high-LET radiations, known to be more cytotoxic than low-LET X-rays, would produce multiple DNA damages that can be overlapped, the method of counting number of foci alone can distort biological risk. Earlier studies investigated the size of foci in connection with the linear energy transfer (LET) of radiation, leaving the issue open to further discussion [3, 4]. In this study, we checked the linearity of DNA damage to radiation dose, and compared the size distribution of gamma-H2AX foci in relation to radiation quality to investigate the complexity of DNA damage. The size of single focus (X-ray) and overlapped foci (alpha particles) were compared with simulation results particularly at low doses.

II. Materials and Methods

A. Cell preparation

The normal human lung epithelial cells (BEAS-2B, CRL-9609, ATCC, VA, USA) were chosen for this study. The cells were grown in LHC-9 medium (12680-013, Thermo Fisher Scientific) with phenol red indicator (PCS-999-001, ATCC) at 37 °C with 5% CO_2.

We used Mylar-bottomed dishes because the alpha particle cannot penetrate the ordinary cell dish. Cells were seeded on Mylar dishes one day before irradiation.

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B. Irradiation

Alpha particle exposure was made by using the SNU-ALPHACELL irradiator containing a 3.793 MBq Am-241 disc source [5]. X-rays were irradiated in the SNU-HARDX facility where an YXLON 450-D08 beam tube generates bremsstrahlung X-rays [6]. The beam tube operated at 150 kVp and 0.36 mA. Cells were exposed at doses of up to 1 Gy at 0.356 Gy/min with both alpha particles and X-rays.

C. Gamma-H2AX assay

Cells were detached immediately after irradiation. Among them, 30,000 cells were attached on microscope slides by using cyto-centrifuge (Rotofix 32A, Hettich). Hydrophobic barrier lines were drawn using PAP pen, to prevent the loss of treatment reagents.

Attached cells were washed by PBST, the Dulbecco's phosphate buffered saline (DPBS) (17-512F, Lonza) with 0.05% tween 20 (T9100-010, GenDEPOT), and fixed with 4% paraformaldehyde (163-20145, Wako) for 5 minutes. Fixed cells were washed twice and permeabilized using 1% Triton X-100 (T9500-010, GenDEPOT) solution for 10 minutes. After three times of washing, samples were blocked with 10% bovine serum albumin (BSA) in PBST. Those cells were treated with anti-gamma-H2AX phosphor S139 antibody (Ab2893, Abcam) in 1% BSA solution. Cells were washed three times, and treated with goat anti-rabbit IgG H&L (Ab6717, Abcam) in 1% BSA solution. Finally, samples were washed three times with PBST and distilled water, and stained with 4' 6-diamidino-2-phenylindole (DAPI) in fluoroshield mounting medium.

Gamma-H2AX signal was identified in a form of several foci by using fluorescence microscope (BX53F, Olympus). The number of foci was counted using a free software 'CellProfiler' (Broad Institute's Imaging Platform). The number and size distributions of foci were recorded. The software recognized the length in a unit of pixel (1 pixel = 0.25595 μm).

D. Montecarlo simulation

The 'Particle and Heavy Ion Transport code System' (PHITS, Version 3.02) was used to obtain dose distributions micron-sized masses. Cell nuclei were considered as 10 × 10 × 5 μm hexahedron mesh filled with water. The specific energy values were recorded for 1,000 meshes (cells) selected randomly among total meshes. The secondary electrons produced by alpha particles were considered to be locally deposited.

III. Results

A. Number of gamma-H2AX foci per cell

Fig. 1 presents the excess number of gamma-H2AX foci per cell expressed by alpha particle and X-ray exposures. The excess numbers of foci induced by alpha particles increased linearly with dose. Each experimental data set fits well a linear function (R-square > 0.96). The slope of linear fitting function is 4.33 and 4.83 with alpha particles and X-rays, respectively.

B. Excess number of foci according to their size

Fig. 2 shows the size distributions of gamma-H2AX foci expressed after alpha particle and X-ray exposures at 0.05 Gy and 1 Gy. The total excess numbers of foci per cell were 0.335 and 0.782 with alpha particles and X-rays, respectively, from the exposure at 0.05 Gy whereas they were 4.05 and 4.82 with alpha particles and X-rays, respectively, from the exposure at 1 Gy (see Fig. 1).

From the exposures to alpha particles and X-rays at 0.05 Gy, the excess number of foci was not significant at 95 % confidence level in all of size groups (Fig. 2-a). Also, the size distributions from alpha particle and X-ray exposures were not significantly different ($p > 0.05$) from each other. The counted foci in X-ray irradiated samples were concentrated in the groups of small sizes (0.205~1.29 μm^2), although the deviation is large. On the other hand, the foci in alpha particle irradiated samples showed relatively even distribution.

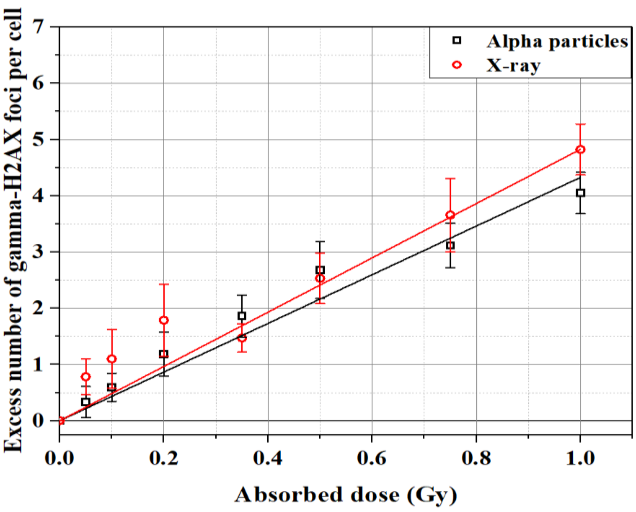

Figure 1. Excess number of gamma-H2AX foci in BEAS-2B irradiated by alpha particles and X-ray.

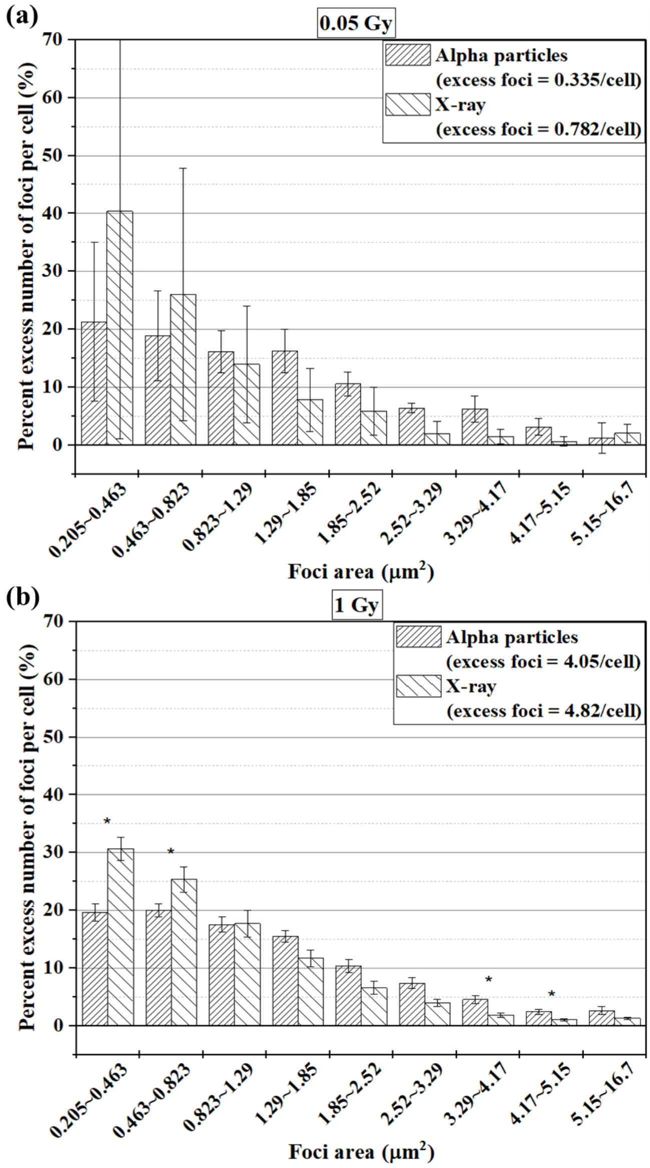

Figure 2. Size distributions of gamma-H2AX foci induced by (a) 0.05 Gy and (b) 1 Gy of exposures to alpha particles and X-rays. The asterisks represent the groups which is significantly different between alpha particle and x-ray exposures (p-value < 0.05).

The exposures at 1 Gy resulted in a significant increase of foci expressions in all of size groups (Fig. 2-b) with both alpha particles and X-rays. The largest excess numbers were counted in the smallest group, and the numbers decrease gradually with an increasing size of foci. The percent excess numbers of gamma-H2AX foci were significantly different at four size groups between the exposures to alpha particles and X-rays (Fig. 2-b). The average sizes of foci from alpha particle and X-ray exposures were significantly different at 1 Gy exposure ($p<0.05$) (Table I).

TABLE I. MEAN SIZES OF GAMMA-H2AX FOCI FOR DIFFERENT DOSE AND RADIATION QUALITY

Dose	Mean size of gamma-H2AX foci (μm^2)		p-value
	Alpha particles	*X-ray*	
0.05 Gy	1.645±0.103	1.321±0.161	0.148
1 Gy	1.577±0.096	1.138±0.055	0.004

C. Specific energy in cell-sized meshes

Fig. 3 shows the energy deposition in cell-sized meshes by exposures to alpha particles and X-rays at 0.05 Gy. Specific energies from X-ray exposures are almost constant around 0.05 Gy. For the same average dose of 0.05 Gy with alpha particles, specific energies are distributed in the vicinity of 0.225 Gy with large variation. The hit probability was approximately 0.25 with alpha particles and 1 with X-ray exposure.

D. Discussion

According to Fig. 2-a and 3, the foci produced by 0.05 Gy of alpha particles have sizes of up to about 1.85 μm^2. It can be assumed that DSBs along the single alpha particle track appear as a focus of about 0.205~1.85 μm^2. Although the foci produced by X-rays are slightly biased to smaller groups as compared to those produced by alpha particles, the difference is not significant (Fig. 2-a).

The size difference according to LET in our study is not as clear as reported in earlier studies [3, 4]. One of the reason is the direction in which cells are exposed to alpha particles. In earlier study, observers looked down the cells in parallel to the vertical beam direction. On the other hand, we detached the cells from Mylar dishes after exposure and then attached them on microscope slides. Most of the cells might have been attached on the slides in different orientations from during exposure to alpha particles. Consequently, the overlapped foci images in former studies might have been separately counted in this study. Nevertheless, the size difference identified at 1 Gy may be due to the presence of some overlapping foci, or the inseparable complex DSBs produced by high-LET alpha particles. Further study shall be performed with regard to foci size difference.

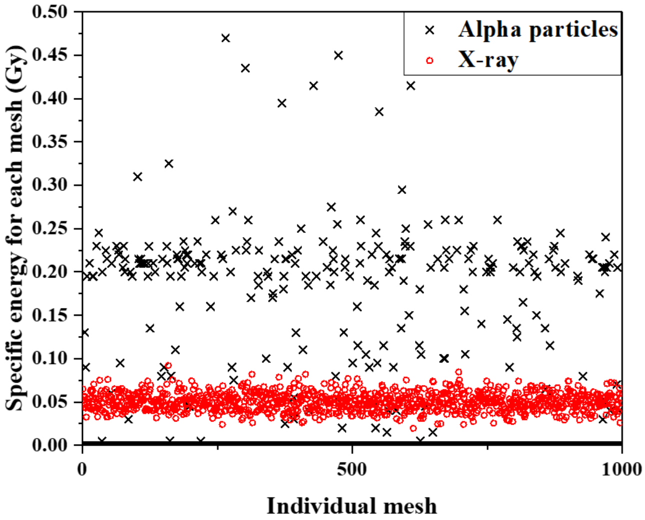

Figure 3. Calculated specific energies for 1,000 micron-sized meshes irradiated by alpha particles and X-rays at 0.05 Gy.

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Toxic Metal Management with Functionalised Titanium Dioxide

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I. Introduction

The mining of metals, including radioactive uranium, is necessary for a variety of manufacturing processes and nuclear fuel production [1]. The impact of this on the environment, however, can be profound. Of particular concern is acid mine drainage (AMD), which is caused by the exposure of sulfur (1) or sulfide minerals (2) to an oxidising atmosphere and large quantities of water, leading to the formation of sulfates and H^+ [2]. The subsequent drop in pH leads to hydrolysis of other minerals and increased solubility of contaminants, including heavy metals, which can pollute surrounding aquatic supplies [1, 2].

$$2S_{(s)} + 3O_{2(g)} + 2H_2O_{(l)} \rightarrow 2SO_4{}^{2-}{}_{(aq)} + 4H^+{}_{(aq)} \quad (1)$$
$$2FeS_{2(s)} + 7O_{2(g)} + 2H_2O_{(l)} \rightarrow 2Fe^{2+}{}_{(aq)} + 4SO_4{}^{2-}{}_{(aq)} + 4H^+{}_{(aq)} \quad (2)$$

This is a naturally occurring process, but is exacerbated and accelerated by mining due to the disruption of mineral deposits [3]. The level of metal contamination in the surrounding environment due to AMD is dependent on the presence and quantity of each heavy metal in the surrounding rock, their reactivity and solubility and the water flow rates and pathways in the area [4]. This contamination poses a high risk to the surrounding environment and human health [5].

Various methodologies have been developed and employed in the task of extracting and separating heavy metals and radionuclides from waste, as well as remediating the acidity of AMD. Adsorption in particular has been shown to be quite economically feasible and effective, even with low metal concentrations [6]. Due to the possibility of desorption, adsorbents are also potentially able to be regenerated and reused, making this a more sustainable option [5]. During adsorption, the metal in the liquid medium interacts with the solid surface of the adsorbent, allowing extraction of the target metal from solution. The adsorption capacity of the material depends on its physical characteristics, such as size, shape, porosity, surface area, presence and reactivity of functional groups, stability and overall charge, as well as the conditions under which adsorption is taking place, including pH, temperature, competing species and contact time [6].

Titanium dioxide (titania, TiO_2) is a versatile material that is cheap to produce and compared to other materials has superior hydrolytic and radiolytic stability [7, 8], which makes it ideal for use in the low pH environment associated with AMD. While metal oxides do have an affinity for heavy metals, functionalisation or modification of the surface can be carried out to increase the extraction capabilities of these materials, and provide additional selectivity. Functionalisation is based on the principle of monolayer assembly, where a series of ligands self-assemble to surround the particle in a monolayer [9]. Ligand design in this study was based on previous work by Veliscek-Carolan et al. (2013), which utilised ligands containing three main structural elements. A terminal alkene was used for addition to the TiO_2 surface, an oxygen or nitrogen containing group was used as the functional aspect of the ligand, as they are capable of binding the target metals, and an extended alkyl chain connecting these two elements was present in order to induce self-assembly on the metal oxide surface [7].

The aim of this project was to optimise the sorption of toxic metals from sulfate media by functionalised titanium dioxide. In order to do this, the titania was synthesised and functionalised with various ligands, incorporating functional groups including amines, phosphates and phosphonates, carboxylic acids and amides, to determine the most effective and selective groups for sorption of toxic metals and radionuclides from sulfuric acid. This work will form the basis for continued research into functionalised titania materials for use in remediation of wastewater and recycling of valuable metals or radionuclides.

II. Methods

A. Synthesis of Titanium Dioxide Particles

Titanium (IV) isopropoxide and glacial acetic acid were mixed for 30 s, then MilliQ water was added, giving a white precipitate. The solid was broken up and left to stir for 30 min at room temperature (r.t.). The mixture was left to settle, liquid decanted, and solid washed with MilliQ water. After washing and adding MilliQ water, nitric acid was added while stirring. The mixture was heated at 70°C for 24 hours and left to settle. The stable sol was decanted into a petri dish and left at 30°C to evaporate. The resulting solid was ground and calcined at 400°C to remove organics, affording titanium dioxide particles.

B. Synthesis of Organic Ligands

The general process of synthesis of the organic ligands is shown in Fig. 1. Organic ligands were synthesised from commercially available reagents (X) containing an undecene chain, which then underwent functional group transformation to give a protected form of the desired organic ligand (R(PG)). This protected molecule was used to functionalise the titania, to prevent the ligand attaching upside down, and then deprotected to give the free functional group.

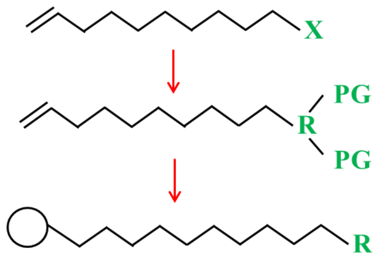

Figure 1. Schematic of the synthesis of organic functionalised titania.

The commercially available 1,8-octanediphosphonic acid ligand was also used, which was attached to the titania via one of the terminal phosphonate groups rather than an alkene.

C. Functionalisation of Titanium Dioxide

Alkene functionalisation: Dried titanium dioxide particles were mixed with a dried and degassed solution of each ligand dissolved in an appropriate solvent with a high boiling point and heated for 24 hours at 110°C under nitrogen. Particles were then filtered, washed and air dried.

Phosphonate functionalisation: Titanium dioxide particles were mixed with a solution of the phosphonate ligand in methanol for 16 hours at r.t. Particles were then filtered, washed and air dried.

D. Sorption Experiment

Mixed stock solutions containing 1 ppm of 16 metals (Na, Mg, Al, K, Ca, V, Cr, Mn, Fe, Co, Cu, Zn, As, Eu, Pb, U) were prepared in sulfuric acid. pH was adjusted to pH 2, 3, 4 or 5 using KOH. Sorption experiments were carried out in triplicate, with 2.0 mL of 1 ppm metal solution at each pH added to 0.01 g of TiO_2. Mixtures were shaken at r.t. for 24 hours. After sorption, each sample was filtered through an individual 0.45 µm syringe filter and the resulting solutions, as well as the stock solutions, were analysed by ICP-MS. Samples were diluted by a factor of 10 in 3% nitric acid before analysis.

III. Results and Discussion

A. Structure of Organo-Functionalised Titania

The final structures of each of the ligands once attached to the titania is represented in Fig. 2. Terminal groups included an amine (NH_2), butyl (DBP) and ethyl (DEP) phosphate esters, a commercially available diphosphonate ($(PO_3)_2$), an iminodiacetate (IDA), an acetamide phosphonate (Ac-Phos) and a hexapeptide (Peptide). These ligands were utilised and synthesised based on literature data and methods in order to assess the affinity of each kind of functional group or combination of functional groups for particular metals and their capacity for extraction. Non-functionalised (NF) titania was also used in the sorption experiments for comparison.

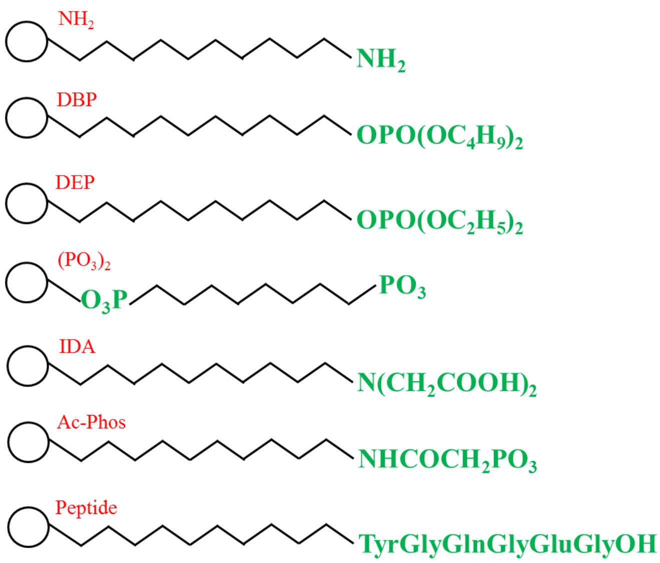

Figure 2. Structures of the commercial or synthesised ligands used to functionalise titania for toxic metal extraction from sulfuric acid.

B. Sorption

Sorption results were determined by ICP-MS analysis, which enables measurement of the concentration of each metal before and after sorption, and thus the calculation of the amount of each metal sorbed as a percentage of the control value. Overall the highest levels of selective sorption occurred at pH 3 and 4. At pH 2, the acidity limits sorption and at pH 5, solubility is lower for many of the metals but sorption increases due to the higher pH.

At pH 2, sorption was limited for most elements, which is as predicted due to the increase prevalence of positively charged hydrogen ions. The hydrogen ions can protonate charged ligands, reducing their attraction to the positively charged metals, as well as compete with the metal ions for sorption sites. Lead was removed at a high level, even at this low pH, by all of the tested ligands. The insolubility of lead sulfate is also a potentially contributing factor in the high level of lead extraction, as precipitation could be stimulated by the presence of the titania particles. There was also some selective sorption of europium and uranium over the lighter elements, with all functionalised titania sorbing a higher percentage than the non-functionalised titania.

At pH 3 there was evidence of selective removal of particular metals, as well as selectively high sorption by some ligands (Fig. 3). Sorption of europium and uranium stayed at a similar level of around 30% by all ligands, while sorption of lead increased from around 60-70% at pH 2 to >90% at pH 3. Notable increases in sorption occurred for vanadium and arsenic, which were sorbed at a high level by the amine functionalised ligand (NH_2), with 52% extracted for both metals. Compared to the sorption by the non-functionalised titania (NF), which was 22% and 17% respectively, this indicates the amine group has an affinity for these two metals.

At pH 4 the selective sorption of arsenic and vanadium continued, with the amine functionalised ligand sorbing 85% and 59% respectively compared to 48% and 27% by the non-functionalised (Fig. 4). Some selectivity was also shown for chromium, with the amine sorbing 26% compared to the non-functionalised sorbing only 3%. The sorption of iron increased substantially, with a max sorption of 32% by the diphosphonate ligand at pH 3 and 73% by the amine at pH 4. Results between different ligands for these were within error, so there was no obvious selectivity by any particular functional groups. An interesting result was some selectivity by the diphosphonate group for uranium at pH 4, with 55% sorption, with the next highest being 35% by the amine. Sorption for lead decreased from pH 3, but the original concentration was also quite low, with only 0.3 ppm in the control solution.

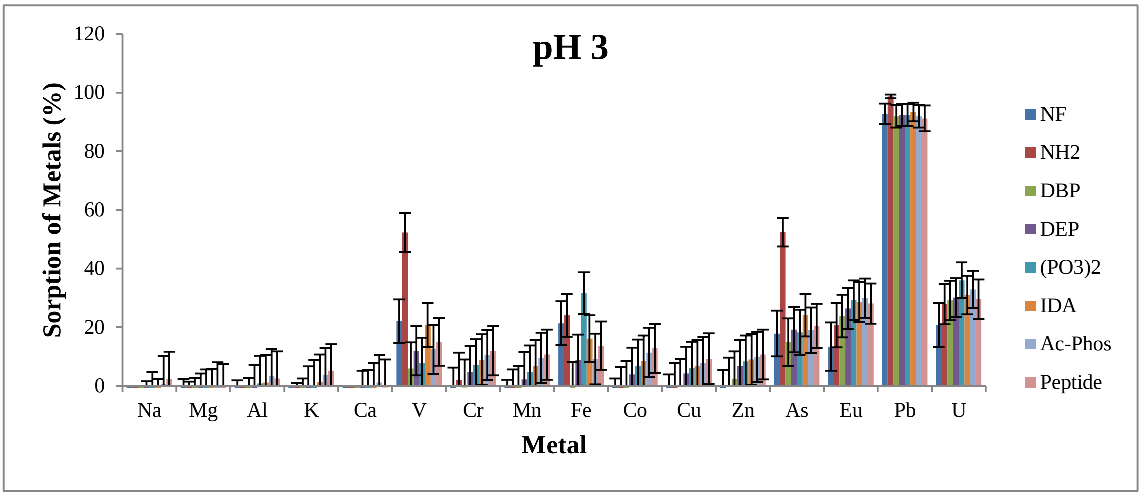

Figure 3. Sorption of metals from sulfuric acid at pH 3 by functionalised titanium dioxide materials, as a percentage of the control. Data is presented as mean ± error (n = 3).

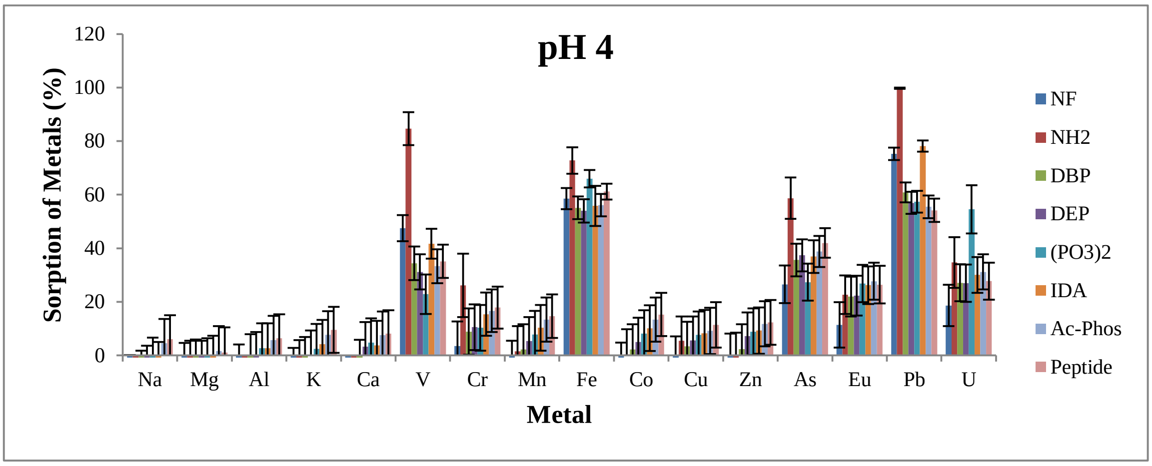

Figure 4. Sorption of metals from sulfuric acid at pH 4 by functionalised titanium dioxide materials, as a percentage of the control. Data is presented as mean ± error (n = 3).

At pH 5, there was a high level of sorption of aluminium, vanadium, chromium, iron, arsenic, lead and uranium, though starting concentrations were often lower at this pH. There was no selectively high sorption of any of these metals by a particular ligand, with all showing similar results. However, the amine and diphosphonate ligands which had shown the most selectivity had not been tested at pH 5 due to a lack of material.

IIII. Conclusions

These preliminary results show promise for the selectivity of the amine functionalised ligand in removing toxic metals such as arsenic and lead from a mix of metals in sulfate media. These materials also showed consistent overall selectivity by sorbing heavier elements at higher percentages than lighter ones, showing the potential to remove toxic heavy metals or radionuclides from sites affected by AMD without removing naturally occurring and widespread lighter elements such as sodium. Further development of these materials to increase surface area, capacity and selectivity through creating more porous materials and new ligands to functionalise with will allow for more effective removal of toxic metals. In the long term, developing a high surface area material with a porous support that can be implemented at mining sites to immediately remediate AMD, would enable prevention of widespread environmental contamination and health issues and allow the mining industry to manage its waste in an efficient and safe manner.

Acknowledgment

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Enhancing Material Properties and Radiotracers by Deuteration

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I. Introduction to Deuteration

Deuterium (^2H) is a stable non-radioactive isotope of hydrogen (^1H) with 0.02% natural abundance. Deuteration is the process by which deuterium is incorporated/enriched within molecules. Deuterium differs from hydrogen by the addition of an extra neutron, and this doubling in the atom's mass leads to marked differences in physical and chemical properties that can be exploited for a variety of applications, including but not limited to:

- Slowing the rate of oxidation and degradation of materials under ambient or extreme conditions.
- Tuning intra- and intermolecular interactions (modifying the hydrogen bonding network to alter the stability, reactivity, and self-assembly of materials).
- Altering bond strength (frequency of vibrations), electronic structure (transitions/excitations), density, and refractive index.
- Internal standards in analytical chemistry (mass spectrometry (MS) quantification of biological analytes).
- Probing fundamental material structure, interactions, and behaviour using analytical techniques (e.g. neutron scattering, nuclear magnetic resonance (NMR) spectroscopy, infra-red (IR) spectroscopy).
- Improving the biological stability of molecules in living systems (slowing the rate of enzymatic breakdown by the kinetic isotope effect) or forcing a switch to another metabolic pathway.

Beyond these niche fields, deuteration remains broadly untapped in science, industry and society. The adoption of deuteration technology in everyday life is expected to lead to new opportunities in high tech materials, advanced manufacturing, personal care products, and health/medicine/diagnostics. There is significant interest in both the progress of deuteration chemistry and applications of deuteration, evidenced by a number of reviews in the last decade [1-3], including back to back reviews in 2018 [4, 5].

II. Deuterated Drugs and Radiotracers

The pharmaceutical industry continues to examine the deuteration of drugs [2, 6-9] to increase metabolic stability. This is important to improve drug efficacy and/or dosing (e.g. lengthen interval between doses, or decrease doses). In April 2017, the U.S. Food and Drug Administration approved the world's first deuterated drug, deutetrabenazine (Fig. 1), which is used for the treatment of chorea associated with Huntington's Disease [10]. Additional deuterated drug candidates in clinical trial stages for other diseases have been recently highlighted [11]. Selected examples include dextromethorphan-d_6 (several phase 2 studies, entered phase 3 studies [12-14]) and pentoxifylline-d_5 (completed a phase 2 study [15]) (Fig.1). The pharmaceutical company Retrotope [16], supplements the human diet with deuterium-stabilised lipid drugs to prevent cellular damage and recover cellular function damaged by lipid peroxidation due to oxidative stress, and has been clinically tested in Friedreich's ataxia.

A type of diagnostic drug, known as radiotracers, enables the medical imaging of diseases. These are designed to target particular biological processes or proteins implicated in diseases, and are routinely used in hospitals to diagnose diseases and track progression. Radiotracers contain a radionuclide which acts as a beacon; the radioactive decay is detected and provides location-accurate information to allow an image the diseased tissue to be constructed. For example, positron emission tomography (PET) enables imaging by detecting the coincident gamma rays arising from annihilation of a positron-emitting radionuclide such as fluorine-18 or carbon-11. To obtain informative medical images, a sufficient amount of the administered radiotracer must remain intact, and metabolites not detract from the location accuracy in the image of the diseased site. A small number of deuterated radiotracers are reported in the literature [17-26], and incorporate deuterium at positions susceptible to metabolic attack, to exploit the kinetic isotope effect (whereby the breaking of stronger carbon–deuterium bonds occurs at a slower rate than carbon–hydrogen bonds), leading to:

- increased availability at the target [17, 18],
- resistance to oxidation [17, 18, 22],
- higher signal-to-background contrast and detection sensitivity of PET [18],
- reduced skull bone uptake of radioactivity [19, 23],

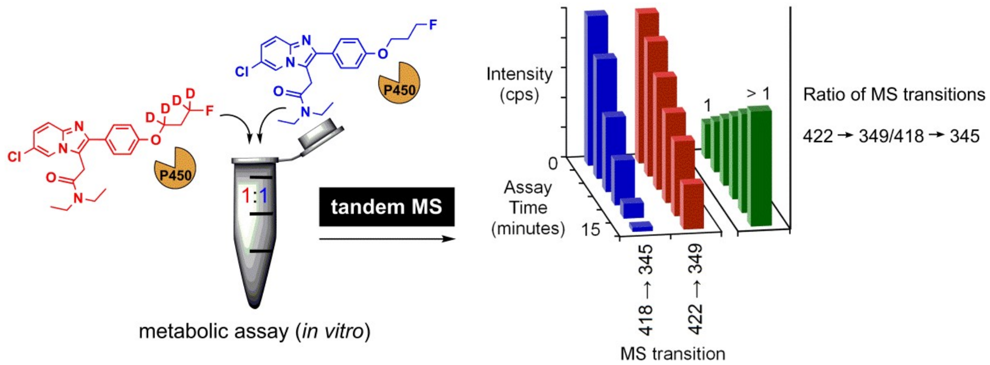

Fig. 2. Schematic of the method to determine the difference in metabolic stability between deuterated and non-deuterated isotopologues, as applied to the radiotracer PBR111 [27]; (a) metabolic assay of a 1:1 mixture of the isotopologues in one-pot, (b) tandem MS (blue and red bars), (c) ratio of the MS transitions of the isotopologues with time (green bars) was calculated, (d) deviation of the ratio from the initial value with time indicated a deuterium kinetic isotope effect.

- reduced trapping in tissues [20, 24],
- reduced clearance favouring a particular metabolic pathway [21, 25, 26].

III. Deuterated PBR111 and a MS/MS Method to Assess Improved Metabolic Stability

PBR111 is an ANSTO-developed radiotracer for the imaging of neuroinflammation [27-30], but is rapidly metabolised at the aliphatic propyl side chain containing the fluorine radiolabel, eventually yielding free fluorine, which is taken up by bone [29]. Deuteration of this propyl motif was undertaken in an effort to improve metabolic stability.

While knowledge of the metabolic profile of compounds can assist in selecting locations to be deuterated, it is difficult to predict whether this will translate to a metabolically favourable outcome [31]. Indeed, there are examples when there is no difference between deuterated and non-deuterated radiotracers *in vitro* or *in vivo* [32], which would be desirable to ascertain prior to undertaking radiochemistry, *in vivo* biodistribution studies and PET imaging. Thus a rapid, simple, and robust method is required so that this can be assessed following compound synthesis, at the *in vitro* stage, and using stable isotope cold standards of the radiotracer.

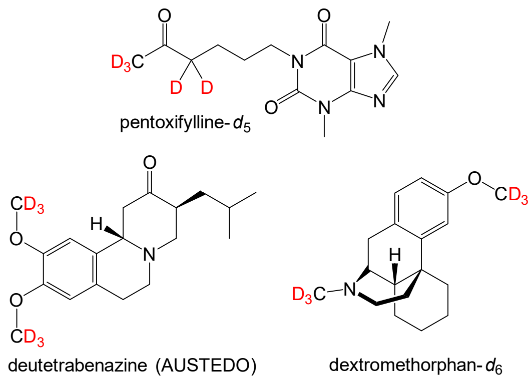

Fig. 1. Selected deuterated drugs at various stages of approval and clinical trials [33].

In preliminary *in vitro* studies, evaluating the difference in metabolism between deuterated and non-deuterated analogues is typically performed separately for both isotopologues (e.g. non-deuterated and deuterated) because the physical chemistry of both compounds is essentially identical and they cannot be distinguished by methods involving liquid chromatography or radiolabelling.

A new screening method was developed [33], which combines an *in vitro* metabolic assay (liver microsomes, which contain drug metabolising enzymes) with tandem mass spectrometry to determine if the site of deuteration has resulted in an improvement to metabolic stability, in a one-pot approach. As the mass (unique to each isotopologue) was monitored, there was no need to analyse the compounds independently. A 1:1 ratio of both the deuterated and non-deuterated molecule was subjected to metabolism, and the relative rate of consumption of both isotopologues was determined by using MS/MS transitions unique to both molecules. A deviation of the ratio of the MS transitions from the initial starting point (e.g. pre-metabolism) with time indicates a kinetic isotope effect. The process is shown in Fig 2.

A. Organic Synthesis

Two deuterated analogues of PBR111 were synthesised (Fig. 3) [33]; PBR111-d_4(propyl) with 99 ± 2 %D on the 1,3-propyl positions, and PBR111-d_4(ring) with 97 ± 2 %D on the phenyl ring. PBR111-d_4(ring) was a control, given previous studies with PBR111 indicated the phenyl ring is not subject to significant metabolism [29], and therefore was expected to be metabolised similarly to PBR111-H. The deuterated PBR111 analogues were fully characterised by NMR, MS, and HPLC.

B. Method

Using a rat liver microsome assay [29], a 1:1 mixture of PBR111-H and PBR111-d_4 was subjected to *in vitro* metabolism over 15 minutes. An AB SCIEX QTRAP 4000 MS/MS with an ESI source was used, with samples (e.g. assay time points, validation standards) directly infused by syringe pump. The dominant analogous transition in multiple reaction monitoring mode was used to calculate the ratio between deuterated and

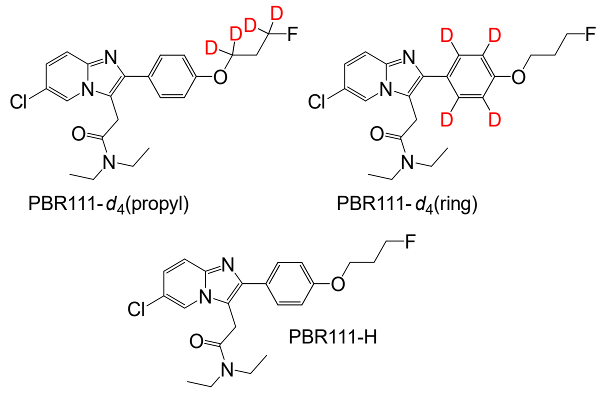

Fig. 3. PBR111-H, and deuterated analogues PBR111-d_4(propyl) and PBR111-d_4(ring) [33].

non-deuterated: m/z 422→349 for PBR111-d_4 and 418→345 PBR111-H, corresponding with intact molecule and the MS-induced fragment for loss of $N(CH_2CH_3)_2$ from the amide pendant to the imidazo[1,2-*a*]pyridine ring. The mean of the ratio for the assay time points was calculated from triplicate assays, with each individual assay time point also measured in triplicate. The full method, including specific details, are reported in [33].

C. Validation

The analysis of known prepared ratios of PBR111-d_4(propyl)/PBR111-H (90:10, 80:20, 70:30, 60:40, 55:45, 50:50) confirmed excellent linearity ($R^2 \geq 0.999$) with the ratio of the counts per second (cps) of the analogous MS/MS transitions for PBR111-d_4(propyl) versus PBR111-H (please see [33]). These solutions were prepared at the same concentrations and in the same media as for the microsome assay, but excluded the liver microsomes. A method blank of the microsome assay media and MS diluent excluding PBR111 confirmed minimal interferences from the sample matrix. The limit of detection (LOD) was ~13 nM and limit of quantitation (LOQ) was ~130 nM for each parent compound.

D. Evaluation

A 1:1 mixture of PBR111-d_4(propyl)/PBR111-H and PBR111-d_4(ring)/PBR111-H were each subjected to *in vitro* metabolism using rat liver microsomes over 15 minutes, and the ratio of the MS/MS transitions calculated.

The ratio of the transitions deviates from 1:1 for PBR111-d_4(propyl) (Fig. 4, green circles), demonstrating improved metabolic stability relative to PBR111-H. To further demonstrate the utility of the method, the ratio of transitions remains relatively constant for PBR111-d_4(ring) (Fig. 4, red squares). Statistical analysis (one-way ANOVA, F-test) supported this data [33]. The results are in line with our previous work on PBR111 [29], and demonstrates the method can distinguish deuteration at a metabolically favourable location. Additionally, pharmacokinetic parameters can be determined from the data [33], and are summarised in Table 1. Using the

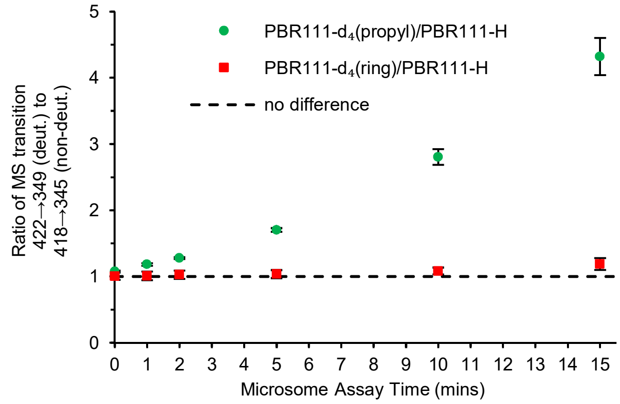

Fig. 4. Relative metabolic stability *in vitro* (rat liver microsomes) of PBR111-d_4 to PBR111-H, for deuteration of the propyl chain and phenyl ring [33]. A positive deviation with time from the starting ratio indicated increased metabolic stability (N=3, error bars are ±1 SE).

TABLE I. SUMMARY OF SELECTED PHARMACOKINETIC PARAMETERS FOR PBR111-H AND PBR111-D_4(PROPYL)

Compound	k (min^{-1})	$t_{1/2}$ (min)	Ratio [a]
PBR111-H	0.27	2.57	3.88
PBR111-d_4(propyl)	0.18	3.85	

[a] Ratio of quantity of PBR111-d_4(propyl) to PBR111-H remaining after 15 minutes based on their respective half-lives, starting from an initial quantity of 2.09 µgmL^{-1} (5 µM) of PBR111-H and 2.11 µgmL^{-1} (5 µM) PBR111-d_4(propyl), calculated using the half-life exponential decay equation, $N_t = N_0 \left(\frac{1}{2}\right)^{\frac{t}{t_{1/2}}}$.

calculated half-life, it can be estimated that there would be 3.88 times as much PBR111-d_4(propyl) remaining after 15 minutes relative to PBR111-H, similar to the ratio of the transitions in Figure 4.

IV. CONCLUSIONS

Deuteration of the radiotracer PBR111 at the fluoropropyl motif increased *in vitro* metabolic stability. An *in vitro*/tandem MS method provided simple and rapid assessment of the impact of the chosen site of deuteration on metabolic profile, and is expected to be able to be applied to any pair of deuterated and non-deuterated isotopologues, including pharmacologically active molecules, to aid in determining the suitability of the chosen site of deuteration. Additionally, for radiotracers such as PBR111, radiolabelling is not required and so stable isotope cold standards can be used for preliminary screening.

ACKNOWLEDGMENTS

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A Simulation Study on Detecting Fake Gold Bar Using Neutron Irradiation-Based Analysis

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I. INTRODUCTION

Gold bars have been used as a currency for the following reasons. It is easily transportable because it has a high value to weight ratio. In addition, it is fungible with a low margin between the prices to buy and sell. But it is well known that fake gold bar exist in the gold market. Fake gold bars on the market cannot be identified easily without testing because they have the same appearance as a pure gold bar. Thus, a non-destructive monitoring method is needed to avoid the trading of fake gold bars on the market.

In the present study, fake gold bar detecting methods that use neutron irradiation-based analysis were suggested. The first detecting method is to measure the target's neutron transmission ratio. The advantages of this method are that it takes less time and it can reduce the activation problem of gold bars. The second detecting method is to use Prompt Gamma Activation Analysis (PGAA). PGAA is an established nuclear analytical technique for the non-destructive determination of elemental and isotopic compositions. To verify the effectiveness of both methods, Monte Carlo calculations of the neutron transmission ratio and the cold neutron-induced prompt gamma-ray spectra of pure and fake gold bars were conducted.

II. METHODS

The neutron transmission ratio and the cold neutron-induced prompt gamma-ray spectra of pure and fake gold bars were calculated by using the Monte Carlo N-particle extended code package MCNPX. This cod package enables one to simulate the transport of neutrons, photons and electrons in the medium and to define three dimensional geometries in an arbitrary way [1]. Thermal and cold neutrons were assumed as the neutron beam. A fake gold bar is commonly made by filling the bar inside with other substances, particularly tungsten, which has nearly the same density (19.3 g/cm^3) as gold (19.25 g/cm^3), but with a cheaper price [2]. The fake tungsten gold bar was modeled to be a 6-mm-thick tungsten bar plated with 1 mm of gold. The simple geometry of the neutron irradiation system for detecting fake gold bars in the MCNPX simulation is shown in Figure 1. The neutron beam is transported to the analytical sample inside the sample mounting box by a neutron guide tube. The sample mounting box measures 20 $\times$ 20 $\times$ 20 cm^3, which is suitable for accommodating gold bars. The neutron fluence according to the depth of the gold bar was calculated using the F2 surface flux tally. In addition, the neutron transmission ratio at each depth was calculated. To calculate the prompt gamma-ray spectra of pure and fake gold bars, transportation of photons from gold bars through the surrounding media to the detector was conducted. The photon source for the MCNPX input was determined by the number of (n, γ) reactions in the gold bar. The neutron fluence spectra were multiplied by the number density and the energy dependent (n, γ) cross sections for gold and tungsten in the fake tungsten gold bar volume to determine the number of (n, γ) reactions. The energy dependent (n, γ) cross section data for gold and tungsten from the publication database of IAEA (Vienna: International Atomic Energy Agency, 2006) were used [3]. All calculations were carried out using 10^6 particle histories, resulting in a target relative error R less than 1 %.

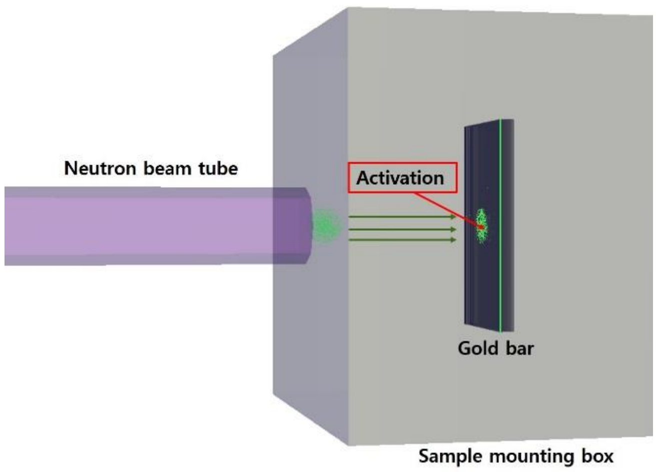

Figure 1. A schematic of the neutron irradiation system for detecting fake gold bars in MCNPX simulation.

III. RESULTS AND DISCUSSION

A. *Neutron transmission ratio of pure and fake gold bars*

The transmission ratios of thermal and cold neutron beams in pure and fake gold bars are shown in Figures 2 and 3. In the Y

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axis of the graph, F_0 and F(x) represent the number of incident neutrons at the surface and the depth of x in the target, respectively. The neutron transmission ratio of the pure gold bar was lower than that of the fake gold bar because the neutron capture cross section of gold is much larger than that of tungsten. In the case of the pure gold bar, 10 percent of thermal neutrons and 0.64 percent of cold neutrons penetrated to a depth of 4 mm. However, in the case of the fake gold bar, 39 percent of thermal neutrons and 8 percent of cold neutrons penetrated to a depth of 4 mm. The transmission ratio of cold neutrons was low compared to that of thermal neutrons. Thus, the cold neutron beam is more suitable for detecting thin gold bars.

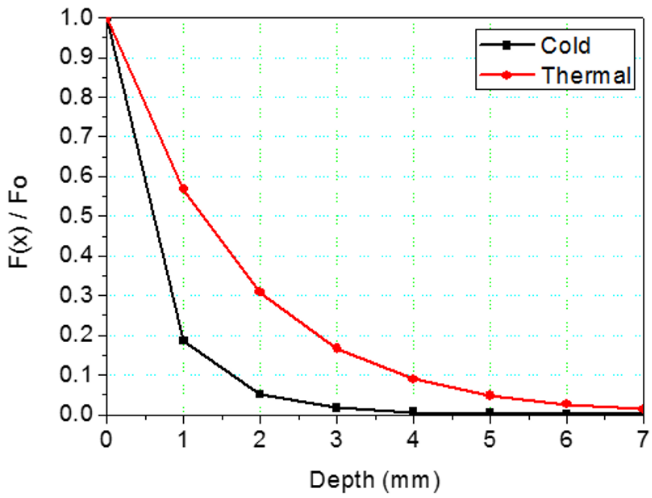

Figure 2. Transmission ratio of cold and thermal neutron beams in the pure gold bar.

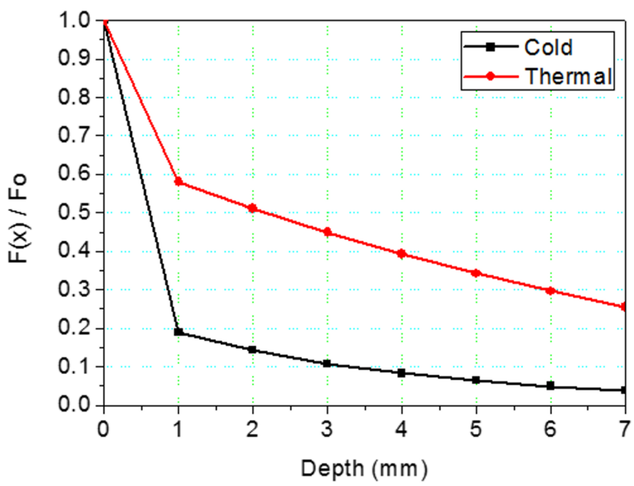

Figure 3. Transmission ratio of cold and thermal neutron beams in the fake gold bar.

B. *Prompt gamma-ray spectra of pure and fake gold bars*

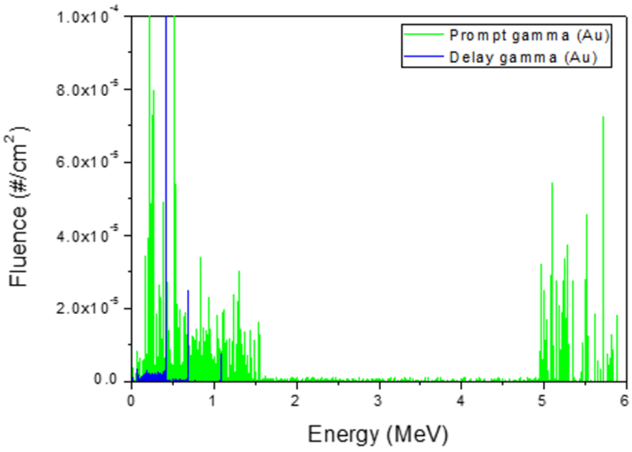

Figure 4. The cold neutron induced prompt gamma-ray spectrum simulated by the MCNPX code for the pure gold bar.

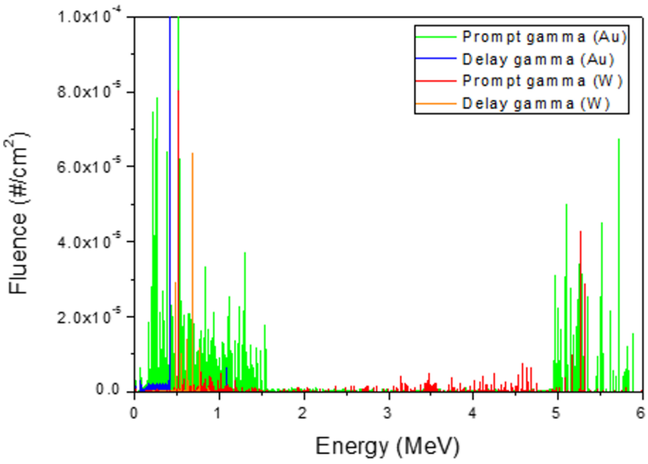

Figure 5. The cold neutron induced prompt gamma-ray spectrum simulated by the MCNPX code for the fake gold bar.

Figures 4 and 5 show the spectra of cold neutron induced prompt and gamma-ray emitted from pure and fake gold bars, respectively. These results indicate the differences in the prompt gamma-ray spectra between the pure and fake gold bars. In Figure 5, the prompt gamma-rays emitted by the activation of tungsten in the fake gold bar were observed clearly. In particular, two gamma-ray peaks of 5.26 and 5.32 MeV showed a high fluence. The total fluence of the gamma rays emitted from the activation of gold was also different, which means that the amount of gold in the fake gold bar is insufficient.

IV. Summary

Monte Carlo calculations of the neutron transmission ratio and the cold neutron-induced prompt gamma-ray spectra of pure and fake gold bars were conducted. In the results, the neutron transmission ratio of the pure gold bar was lower than that of the fake gold bars. In addition, the high-energy gamma-rays from the tungsten fake material and the reduction of prompt gamma-rays from the gold can also be clues for detecting fake gold bars. Therefore, the effectiveness of the methods for detecting fake gold bars is verified.

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From Sketches to Sales: Lessons from the delivery of a nuclear medicine processing facility

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I. Introduction

The ANSTO Nuclear Medicine (ANM) project was announced by the Australian Government in 2012 which included the construction of the first purpose built molybdenum manufacturing facility in Australia, to be situated at ANSTO's Lucas Heights campus in Sydney.

In this paper an overview is given of the journey of completing the ANM facility, with a discussion presented on the delivery approach, innovate design elements and methods of licensing that culminated in the successful transition to commercial operations that commenced in 2019.

II. Brief History

A. ANSTO and Nuclear Medicine

ANSTO is an Australian Government *'Corporate Commonwealth Entity'* that is enabled by legislation to, amongst other things; perform activities *"in connection with the production and use of radioisotopes, and the use of isotopic techniques and nuclear radiation, for medicine…"* [1].

For many years, ANSTO has produced a range of radiopharmaceuticals for Australian and international communities, including a full supply chain supporting sodium pertechnetate [99mTc] based Gentech® Generators through use of production and processing facilities, as well as the OPAL multi-purpose research reactor, all located at the Lucas Heights campus in Sydney/

The molybdenum processing facility in use by ANSTO prior to the ANM project was established a number of decades prior through the repurposing of a series of research hot cells.

III. Creation Of The ANM PRoject

A. Announcement

The Australian Government accounted the investment into the ANM project in 2012, which included funding for two facilities [3]: the molybdenum processing facility and a sister facility to apply ANSTO's Synroc technology to the treatment of liquid waste produced during molybdenum separation (out of scope for this paper).

B. Project Goals

The goal of the project was to triple the production of molybdenum at ANSTO, meaning 3000 6-day Curies[1] was the design target for weekly processing capacity at the ANM facility. At the time, this capacity represented approximately 25% of the global demand [3]. The design life of the facility was set to be 40 years, in harmony with the OPAL reactor lifespan.

Figure 1. The façade of the completed ANM facility

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[1] *'3000 6-day Curies'* is equivalent to 500TBq of Mo99 measured at the end of processing. The *6-day Curie* is a non-SI unit of measurement historically used between reactor operators, Mo-99 producers, generator manufacturers and unit dose preparation facilities. It represents the amount of Mo-99 that will be available 6 days into the future from a given calibration processing or dispatch milestone. The limitations and proposal to replace the *6-day Curie* is discussed in [4].

C. *Management*

As part of the formal project establishment, ANSTO appointed a program manager and formed a project team from within the Major Capital Programs area of the Engineering division who were to manage the works to operational handover

IV. A Phased Design and Delivery

A. *Conceptual and Preliminary Design*

Whilst conceptual design sketches existed as far back as 2007 [5], a facility of this scale and complexity necessitated a phased design approach in order to deliver value and compress the delivery timeline.

In working to complete the preliminary design, ANSTO engaged a domestic top-tier consulting firm to produce a package covering the traditional architectural and building services elements, but retained in-house the design specifications of the Instrumentation & Control, Waste Management, and Radioactive Ventilation components supported by industry consultants.

The implemented facility as shown in Fig. 1, has remained similar to the original architectural vision, including the iconic use of Australian wood integrated into the façade.

B. *Process Design*

The design, procurement and implementation of key molybdenum process elements, which was founded in part from a 2012 intellectual property transfer agreement with the South African company *NTP Radioisotopes* [5], was retained in house under the ANSTO project team.

C. *Design and Construct Procurement*

The single largest procurement from ANSTO was the awarding of an AS4092 'design and construct' [6] contract to a domestic top tier construction firm based on the preliminary design.

That head contractor engaged a domestic top tier design consultant that had access to nuclear design expertise from their North American branch to assist. Under the head contract, many different domestic companies performed a majority of the construction, fabrication and installation work. A key exception were the nuclear hot cell components which were fabricated in North America and installed using their specialist tradespersons.

D. *Directly Managed Procurements*

The process, and some other key items such as the Process Automation and Environmental Monitoring Systems were retained in-house for final design and delivery, utilizing ANSTO or outsourced local fabrication as required.

Additionally, some supply of off the shelf nuclear elements, such as the manipulators and flask solutions were procured and imported from Europe

V. Staffing and Resources

A. *A small team, but large impacts*

The core ANSTO project team throughout the project was on the order of tens of staff, in stark contrast to the more than one thousand tradespersons who were inducted onto site. It is approximated that more than 2500 persons, a clear majority of which were Australian's were involved in some way in the delivery of the ANM facility. This has built clear capacity, both at ANSTO and in local industry.

B. *Graduate Heritage*

A pleasing reflection is the impact ANSTO's Graduate Development Program had on the project. The program takes a cohort of university graduates every two years, allowing rotations through different business areas at ANSTO, with many securing ongoing work. At the peak, the core ANSTO project team included Engineers that had commenced across four successive cohorts and been retained in Project Engineering and Project Management roles within the Major Capital Project group.

VI. Technical Composition of Facility

A. *Overview*

The facility is comprised of three levels. The ground floor contains the main production hot cell suite, supporting hot cells, receipting and dispatch areas, and preparation laboratories. The upper floor contains the plant and instrumentation and control (I&C) rooms as well has access to the top of cells, and a meeting room, whilst the basement contains many filters and tanks for managing process gases, hot cell exhaust and liquid wastes.

B. *Hot Cells*

There are ten hot cells within the facility, six directly related to the manufacturing steps, one for management of process samples, two for storage of solid wastes and for one management and sampling of liquid wastes. There is space and utility connection reserved for expansion for another five hot cells to add a second production line, one of which is currently being delivered to operable in 2020. An image of key hot cells is provided at Fig. 2.

Shielding requirements for the hot cells were set and met as; *"The biological shielding is designed to achieve the following dose attenuation: • Maximum dose rate at contact on front of cell surfaces: 0.003 mSv/h. • Maximum dose rate at contact on rear of cell surfaces: 0.01 mSv/h"* [7].

Dependent on the source term developed, two shielding approaches were designed and constructed. For five of the hot cells, approximately

1.1m thickness of high density concrete[2] was utilized, whilst the others utilized a thickness between 21-27cm of lead encased in 1.3cm of steel to achieve the required shielding.

This shielding approach was shown to satisfy the design objectives after various testing, commissioning and in-use surveys were conducted.

C. Building integration and automation

The whole facility is highly automated, with all elements of the facility systems integrated using a PLC and computerized SCADA based solution. Some operator interfaces and camera displays are visible in Fig. 2.

The scale of the instrumentation and control (I&C) is represented by an input/output (I/O) count of approximately 3500 and implemented using remote I/O on an Ethernet/IP fieldbus. The largest system by count being the Active Ventilation, with an I&C footprint of 12 cabinets. There are 15 human machine interfaces (HMI) supported by dual redundant networks and servers.

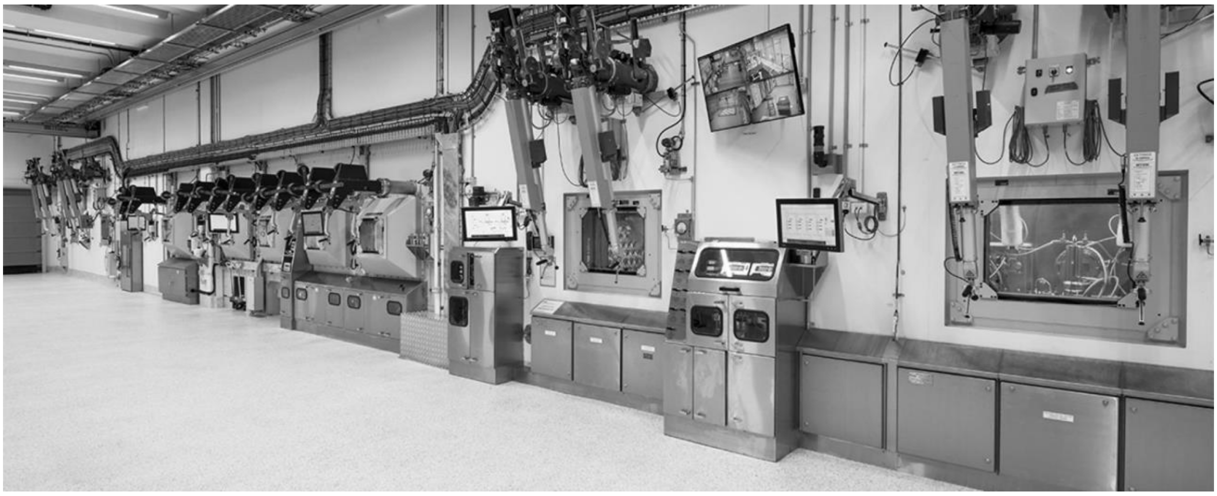

Figure 2. The production cell face located on the ground floor of the ANM facility

Within the I&C systems is a Safety Instrumented System (SIS), build according to the AS/ISO 61511 [8]. 22 Safety Instrumented Functions (SIF) are implemented in the SIS, performing various interlocking functions.

VII. Nuclear Licensing Timeline

The relevant nuclear licensing authority for this facility is the Australian Radiation Projection and Nuclear Safety Agency (ARPANSA).

A. Siting and Construction

In 2013 a siting license was granted, and a construction license granted in 2014 approximately six months after an application was made. A condition on the construction license [9] was that approval was required from ARPANSA for the *"construction of any hot cells/cell containment, hydrogen gas and detection system and any cranes along with items identified as safety category 1 and safety category 2 in the Preliminary Safety Analysis"*. Submissions for these approvals was managed by the core project team with support of ANSTO's Safety, Systems and Reliability staff in line with the project schedule.

B. Operation

In 2018 a license to operate was granted for the facility, which also included a number of conditions [10] that related to the conduct of initial 'hot' commissioning, transitional arrangement between the new and existing facility, as well as elements for particular reporting and review during the first half decade of operations. In 2019, key conditions were lifted which enabled routine operations to occur in the facility. In conjunction with the license granted by the medical regulator, commercial operations commenced soon after.

VIII. Future

The design implementation, including building space and utilities reservation allows for a large degree of flexibility during the design life.

As mentioned in VI.B above, an additional hot cell is currently under construction that will replicate key processing equipment, whilst other spaces exist that will allow the bringing into the facility of various quality control equipment and support steps at a future time.

The flexibility in the facility, including the configurability of the automation system mentioned in VI.C above allows for continuous improvement across safety, environmental and

[2] High density concreted used in the facility was formulated to achieve a density of $3500kg/m^3$, as compared to a typical concrete density at approximately $2400kg/m^3$.

operational, and supports implementing future technologies as they develop over the facility lifetime.

IX. Conclusion

This paper presented a number of regarding to the design, delivery and licensing of the ANSTO Nuclear Medicine facility situated at the Lucas Heights campus in Sydney, Australia. The facility entered commercial operations in 2019, approximately 7 years after the announcement of funding from the Australian Government, and has seen success due to the involvements of thousands of persons across ANSTO, the Australian industry and nuclear specialists from the South Africa, North America and Europe.

Acknowledgment

Congratulations is given to all involved in the delivery of the world class ANM Facility, which has made a lasting positive impact for health in Australia and around the world.

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Pharmaceutical Risk Management for radiopharmaceutical product and process development

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I. Introduction

The use of risk-based approaches is integral to ensuring that the highest quality products are manufactured. Risk assessment should be applied throughout product and process development. Risk assessments can be used to assess product and process safety, product efficacy risks and product quality risks. To develop a product the target quality profile, or intended product use, must be defined. The critical quality attributes must be identified and the critical process parameters recognized, so that adequate control strategies and maintenance strategies can be put in place.

Risk assessment for radiopharmaceutical products, as highlighted by PIC/S Annex 3 Clause 9, "is of even greater importance in the manufacture of radiopharmaceuticals because of their particular characteristics, low volumes and in some circumstances the need to administer the product before testing is completed [1]."

At each stage of radiopharmaceutical product development regulatory submissions must be made both to the local pharmaceutical regulator and the radiation regulator, in Australia they are the TGA and ARPANSA, respectively.

Risk assessments are used to outline the safety of each stage of process development. Risk assessments are essential to product development as, "effective quality risk management can facilitate better and more informed decisions, can provide regulators with greater assurance of a company's ability to deal with potential risks and can beneficially affect the extent and level of direct regulatory oversight" [1].

The aim of risk assessment is to build a product and process with Quality by Design (QbD) as "quality cannot be tested into products" [3]. The same holds true with process safety. The International Society of Pharmaceutical Engineering Baseline Guide for Sterile Products states that "risks associated with a quality product should be identified so they can be mitigated either by engineered solutions or by procedural solutions. [3]". It is imperative that there is quality in both the product and process design.

To design a quality product, the intended use of the product must be defined.

II. Define the intended use of a product

Before the risks to a product or process can be assessed the Quality Target Product Profile (QTPP) must be defined. The QTPP 'forms the basis of design for the development of a product" [3]. The QTPP, or the products intended use, includes parameters such as dose strength, dissolution, intended use etc. for a normal pharmaceutical. For a radiopharmaceutical other parameters such as the radiation dose must be assessed, as the activity of the final solution can limit the batch size.

Risk assessment can be used to define the impact of the API on the target disease, as well as what excipients are most beneficial for the product, in terms of delivery and release. For radiopharmaceuticals the effective radiation dose to the operator when manufacturing the product must be considered, as well as the radiation dose to the patient, the half-life of the isotope and the biological life of the isotope.

III. Critical Quality Attributes

Based on the QTPPs, the Critical Quality Attributes (CQAs) can be determined, as CQAs are "those product characteristics [that have] ad impact on product quality [3]". CQAs must be identified so that they can be scientifically controlled and understood. CQAs are identified and assessed using a variety of risk assessment methods, at different stages of the product development process.

A Preliminary Hazard Analysis (PHA) can be used to identify the CQAs of a product. PHA can be used early in the project when there is little information available to the risk assessment team [2]. CQAs can also be identified through prior knowledge and scientific tests. CQAs impact on the QTPPs can be quantified in a cause and effects matrix [5].

Critical Quality Attributes for a radiopharmaceutical product can be the purity of the product, the composition, the specific radioactivity, the biological activity and the potency. For a radiopharmaceutical the specific radioactivity can change over time, so it is important to factor in the desired activity at the product destination and the half-life of the isotope when defining the CQA.

An example of the CQAs for an oral dosage source is demonstrated in Table III.1, adapted for a radiopharmaceutical. By defining which parameters influence the CQAs, the risks can be kept to a minimum for the product during manufacture.

ANSTO, Lucas Heights, Australia

Table III.1Cause and Effect Matrix for an example oral dosage form adapted from [8]

Critical Quality Attribute	Drug substance				
	Assay	Impurity	Water Content	Residual Solvent	Particle Size
Appearance	Low	No	No	No	No
Assay	High	No	No	No	No
Content Uniformity	Low	No	No	No	Low
Dissolution	Low	No	Low	No	High
Impurity	No	Medium	No	No	No
Residual Solvent	No	No	No	Medium	No
Water Content	No	No	Low	No	No
Specific Activity	High	Low	Low	Low	No

IV. Critical Process Parameters

Once the CQAs have been identified, the design space can be defined. The design space is a concept developed in ICH Q9, as the "relationship between the process inputs … and the critical quality attributes (CQAs)" (ICHQ9 Part 2 section 2.4).

While ICH Q9 is not "intended to create new regulatory requirements" the PIC/S Guide to GMP Annex 3 for radiopharmaceuticals states, "the critical parameters should normally be identified before or during validation and the ranges necessary for reproducible operation should be defined." Annex 3 Clause 36. 'Critical Parameters' mentioned by the PIC/S code are similar to the 'Critical Process Parameters (CPPs) which have impacts on the 'Critical Quality Attributes' (CQAs).

A CPP can usually be measured using a control measure such as time, speed or bacterial count. Once the CPPs have been determined and a control method agreed upon, their control can be designed into the process used to develop the radiopharmaceutical product. The level of control for a required from a CPP can impact the equipment or unit operations selected for a process [5]. Various forms of risk assessment can be used to link the CQAs and CPP, such as FMECA and HACCP.

Failure Modes and Effects Criticality Analysis (FMECA) is a systematic procedure for the analysis of a system to identify the potential failure modes, and their cause and effects on system performance. FMECA can be used to eliminate, contain, reduce or control the potential failures in a system. FMECA can be used to prioritize risks and monitor the effectiveness of the control and is a particularly useful tool when developing a maintenance and reliability plan for a process.

A Hazard Analysis and Critical Control Point (HACCP) study can be conducted. HACCP studies determine the specific hazards in process and develops preventative control measures to minimize the risk of manufacturing defective products. The outcomes from this process are 1. Detection strategies and 2. Prevention strategies. These strategies can be implemented and their effectiveness assessed.

The CPPs for a radiopharmaceutical could be the time taken to manufacture the product, the mass of the product, the pH of the diluting acid. CPPs can be ranked as part of the hazard assessment, which informs the depth of the following risk studies.

V. Critical Control Points

A HACCP study should be conducted at the start of process development using the identified CQAs to determine what the Critical Control Points (CCPs) are. CCPs are process stages where control can be applied, for example the activity of radioisotope that has been dispensed can be controlled using an ion chamber and a gamma spectrometer.

In addition, it is at this point that a risk assessment of the product development that dose rates must be determined to allow for the sizing of the shielding required to protect the process operator, as highlighted by ARPANSA "following a risk assessment, measures to control the radiation exposure of … staff should be established" [5].

VI. Critical Process Equipment and Steps

One essential piece of process equipment is the HVAC system. For the example of a radiopharmaceutical product, the control of the HVAC system must be assessed. HVAC is used to protect the product from airborne pathogens, but also to protect the operator from airborne contamination. HVAC is also a form of barrier to the release of contamination from the production a facility. It is essential that the risks be adequately assessed.

From Annex 3 Clause 26: "For manufacture of radiopharmaceuticals a risk assessment may be applied to determine the appropriate pressure differences, air flow direction and air quality.” This clause highlights the importance of risk assessment for the safety of a radiopharmaceutical product.

This is because the safety requirements for patient safety versus operator safety must be considered. The pressure profile, and leak rate for an installation must reflect the need to contain a radiopharmaceutical source release using a negative pressure barrier. However, the need for a patient safe product requires that the cleanest area for the product manufacture occurs at a 15 Pa or higher pressure than the surrounding areas [4]. Therefore it is essential to carry out a risk assessment to highlight the risks and to design in controls, such as an allowable leak rate, or HEPA filters into the area of lower pressure.

VII. Maintenance

For radiopharmaceuticals, to achieve the desired product output, maintenance must be part of the QbD, as any breakdowns can cause extended delays as they may occur in areas that are too radioactive for maintenance staff to enter. Therefore risk assessment is required during the product and process development phase is to ensure reliability and maintainability of the process. “Preventive maintenance, calibration and qualification programs should be operated to ensure that all facilities and equipment used in the manufacture of radiopharmaceutical are suitable and qualified. [1]” (Annex 3 Clause 20) A good maintenance plan is based off an FMEA study of a system. A good maintenance method is Reliability-centered Maintenance (RCM). RCM is “a process used to determine what must be done to ensure that any physical asset continues to do what its users want it to do” [5]. This study is used to find the failure modes of a system, and how they can be controlled through maintenance tasks, taking into account the frequency of a specific cause of that particular type of failure and its consequence. This stage of risk assessment allows for the maintainability of a product line to be designed into it.

VIII. Conclusion

In conclusion it is clear that the use of risk assessment is integral to developing a product which is safe both for the patient, but also for those who manufacture it. Risk assessment is used to define the Design Space for a product to ensure that it has Quality by Design. Using the design space, the Critical Quality Attributes of a product are defined. Using the knowledge of the CQAs a HACCP or other risk assessment can be carried out to determine the CPPs and to develop strategies to control them. Finally, the effectiveness of these controls can be reviewed using an FMEA or other relevant risk assessment to ensure continued quality within the system. For the manufacture of radiopharmaceuticals, it is particularly important to highlight the critical parameters as there is higher inherent risk in their manufacture. It is clear that risk assessments must be detailed to ensure both patient and operator safety.

Acknowledgment

The author would like to thank those who are reading this, and send them luck in all their future endeavors.

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Maintenance Factors Affecting Increased Lutetium-177 Production

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I. INTRODUCTION

Current forecasts project the production and use of Lu-177 for therapeutic use to be an area of significant growth in the radioisotope market over the coming years specifically in the therapeutic treatment of neroendocrine, prostate and other cancers. With several radiopharmaceutical treatments coming to the end of successful periods of clinical trial, the challenge for manufacturers is to meet this growing demand in a reliable and repeatable fashion.

Naturally, as production capacities increase the plant downtime available for maintenance decreases. Similarly, the potential impact of equipment failures increases dramatically. The key task for engineering and maintenance, therefore, is the critical review of existing maintenance strategies and Computerised Maintenance Management System (CMMS) to ensure failures are, at worst proportional to the associated production increase.

This takes on added significance in the context of radiopharmaceuticals where principles of radiation safety and Good Manufacturing Practice (GMP) apply. Not only does maintenance require a focus on pure production outputs but also on the expectations of two strict regulatory frameworks. The success of the process relies on correct management through a combination of methodologies.

II. MOTIVATIONS FOR A NEW MAINTENANCE STRATEGY

The production of Lutetium 177 commenced in 2014 following the construction of a dedicated production facility. The facility supported Australian clinical trials of novel treatments for neuroendocrine and prostate cancers over the following years. Over time, the positive results of these trials cause an increased demand in Lutetium supply and the production increased from one run per fortnight to weekly runs aligned with the reactor schedule.

This continued support of trials saw demand, particularly in the European markets, increase further driving the decision to aim for two production runs per week by March 2020. However, concerns over the reliability of the process prompted further investigation into the existing maintenance strategies of the production plant.

In addition to the commercial drivers for the review, failure of the vial-crimping unit in March of 2019 highlighted the gap in the maintenance strategy around critical GMP equipment. In particular the lack of appropriate validation and return to service requirements for such equipment. The failure of the crimper threatened to put production down for a minimum of two weeks while an external laboratory performed microbiological testing to confirm the integrity of the vial seal.

Despite the significant implications to production, no dedicated maintenance strategy existed around the vial-crimping unit or its validation requirements. In comparison, validation procedures existed in the maintenance strategy for other critical items such as the autoclave used for terminal sterilisation of the product. This prompted investigation into the allocation of maintenance effort as shown in Fig. 1.

III. DATA ANALYSIS – CLOSING THE GAP

Analysing the maintenance data served to verify the assertion that gaps existed within the existing maintenance strategies. A number of plant functions critical to the process (notably purification and dispensing) showed planned maintenance (PM01) calls contribute less than 40% of total calls. Fig. 1 shows this gap to great effect. The graph also served to highlight issues with the functional and reporting structure whereby maintenance tasks do not fall under discrete functional blocks for trending purposes.

TABLE I. MAINTENANCE CALLS BY YEAR

Year	MD01	PM01	PM02	PM03	PM04	PM05	PM06	PM07
2014	-	66	48	-	1	-	5	-
2015	-	154	78	1	6	2	-	2
2016	-	170	54	-	10	2	1	-
2017	8	129	44	6	-	-	-	2
2018	4	212	49	14	-	-	-	10
2019	1	90	22	4	-	-	-	-
Total	**13**	**821**	**295**	**25**	**17**	**4**	**6**	**14**
Per. (%)	**1.1**	**68.7**	**24.7**	**2.1**	**1.4**	**0.3**	**0.5**	**1.2**

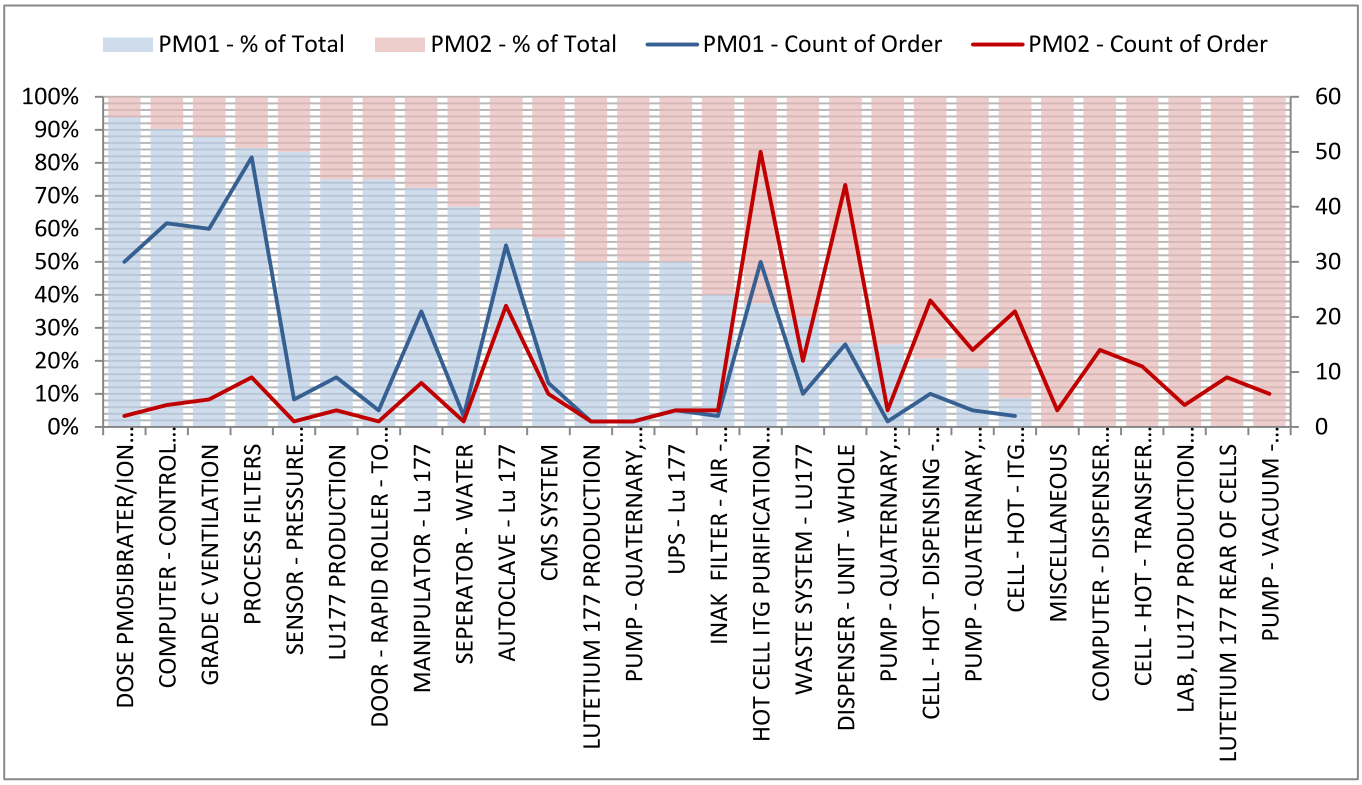

Fig. 1 Lutetium 177 Maintenance Data - Comparison of Planned and Unplanned Tasks

In addition to the assessment of maintenance tasks by equipment, additional investigation occurred into maintenance tasks by year. As shown in TABLE I, unplanned (PM02) maintenance calls average to 50 per year and make up 24.7% of total calls. This again serves to highlight a lack of planned maintenance attention and prompts further review into the nature of the regular rate of unplanned maintenance calls to determine whether there are recurring faults.

IV. Good Manufacturing Process Maintenance

Quality by Design (QbD) is a risk-based design methodology pioneered by the International Conference on Harmonisation of Technical Requirements for Registration of Pharmaceuticals for Human Use. Developed in 2004, ICH Guideline Q8 (R2) on Pharmaceutical Development [1], proposed a "bottom-up" method of product development that prioritises factors affecting patient safety and GMP impact in the very early stages of the development process.

The process begins with a Quality Target Product Profile (QTPP). This forms the specification of the final product with many publications comparing the QTPP to the information on the product label. In the pharmaceutical sphere this is a list of Quality Attribute attributes such as the dosage form, appearance, sterility and content uniformity (among others) and their target values.

These Quality Attributes are then categorised through a risk assessment to identify Critical Quality Attributes (CQAs). The objective to assist the development process by emphasising those attributes that have an impact on product safety.

Quality attributes	**Target**	**Criticality**
Dosage form	Injection	Not applicable
Potency	15 mg	Not applicable
Physical description	Yellowish viscous liquid	Not applicable
Appearance	Clear and not more intense than reference	Critical
Identity	Positive	Critical
Assay	98.5–101.5%	Critical
Impurities	≤0.1% (2,2′-oxydiethanol) Total impurity≤1.2%	Critical
Water	≤0.5%	Not critical API does not hydrolyse
Content uniformity	EP	Critical
Heavy metals	≤10 ppm	Critical
Microbiology	Meets USP if tested	Not critical, a precursor to dissolution

USP: United States Pharmacopeia, API: active pharmaceutical ingredient, EP: European Pharmacopoeia

Fig. 2 Example Quality Target Product Profile [3]

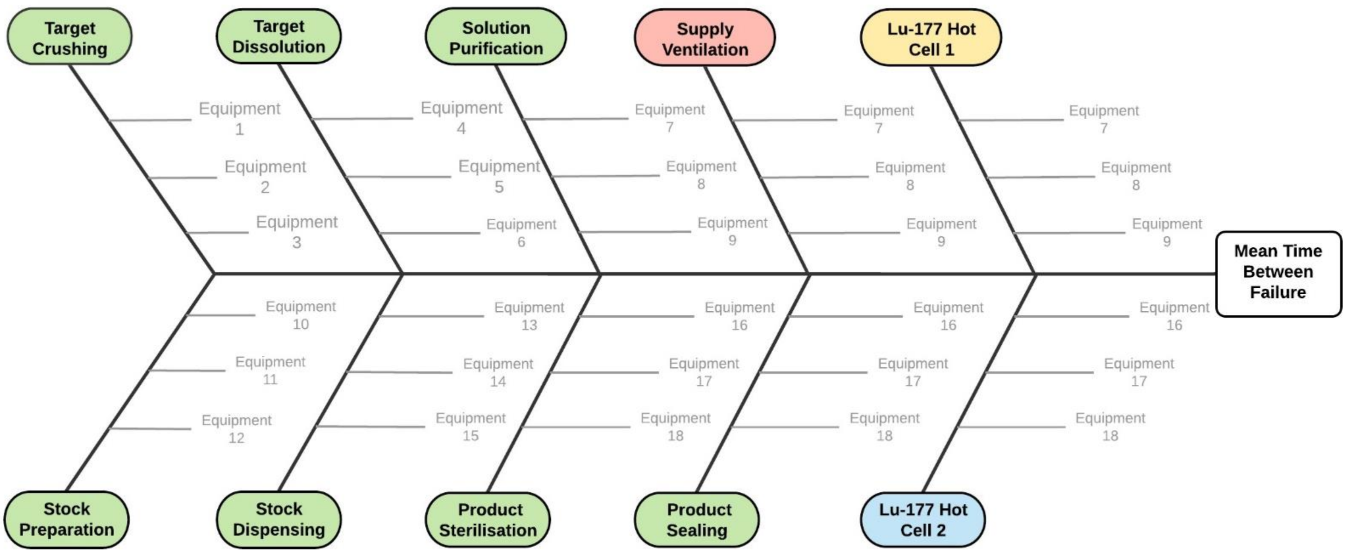

Fig. 3 Example MTBF Ishikawa Diagram Showing Multiple Functions Affecting the Desired Outcome

The process then uses traditional risk management methods such as Failure Mode and Effect Analysis (FMEA), Fault Tree Analysis (FTA) and similar, to identify those parameters that impact the CQAs. These form the Critical Process Parameters (CPPs). CPPs are parameters that, outside a proven or validated range, impact product quality and include items such as temperature, mixing speed, mixing time and similar controllable elements.

The International Society for Pharmaceutical Engineers (ISPE) published an article in their periodical magazine detailing the benefits of QbD observed by several significant pharmaceutical organisations [2]. The key observations highlighted in this report in relation:

- Significant cost savings
- Reduced time to market
- Improved process knowledge and understanding
- Improved hand over from the lab to manufacturing
- Improved quality attributes

The benefits of this process from a maintenance perspective lies in the improved hand over from the development process to the operations and maintenance cycle. In particular the improved understanding of those parameters critical to patient safety. While a model centred on supporting Quality Risk Management (QRM), knowledge of critical process parameters and, more specifically, the functions those parameters affect, allows focusing of strategies around GMP relevant maintenance. As noted previously, review of the existing strategies highlighted a lack of comprehensive effort in this area.

V. Improving Plant Knowledge and Structure

As stated in Section III, assessment of the existing plant data highlighted inefficiencies in the data collection and assignment methodologies. A comprehensive audit of the installed equipment showed that the CMMS lacked information on in excess of 119 unique components. That considers spares data, care and maintenance instructions, service intervals and risk assessment data offering knowledge of failure impacts to system function. As with the issues relating GMP systems, failure of a then-to undocumented pneumatic cylinder function in the fourth quarter of 2019 threatened to put the plant out of production.

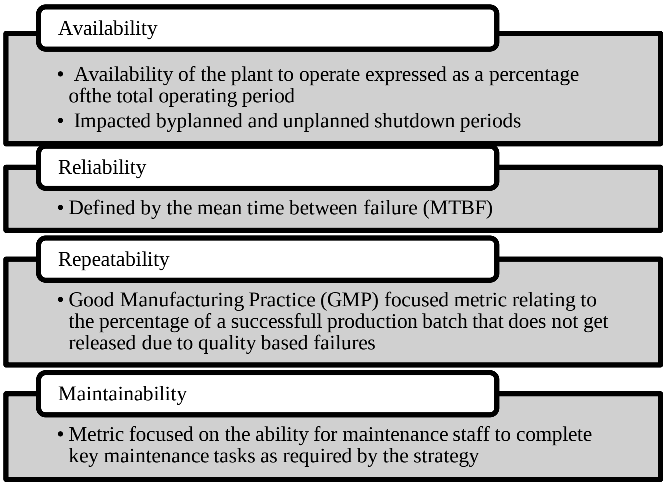

Fig. 4 Production Performance Metrics

The root cause of this missing plant knowledge was the reliance on vendor operating and maintenance manuals for maintenance information. While sufficient in the early stages of production, the failure to evaluate this information in light of operational data and expand on the available information through regular audits and condition monitoring introduced unnecessary reliability risk into the plant.

In addition to the audit, restructuring of the CMMS data based on function and the establishment of corresponding functional locations allows for the implementation Reliability Centred Maintenance methodologies based on the IAEA Guidance document [2]. Coupling the above improved system knowledge and correct respect given to GMP items in Section IV, improved metrics beyond simple reliability and availability.

VI. Defining Performance Metrics

Fig. 4 offers examples of metrics developed to measure the condition of the Lutetium maintenance strategy. While those presented focus specifically around reliability, other metrics relating to net plant utilisation and safety of operation require similar definition.

Borrowing from the concepts of QbD expressed in Section IV, the intention is to link all functions against one or more defined production targets. Fig. 3 Example MTBF Ishikawa Diagram Showing Multiple Functions Affecting the Desired OutcomeFig. 3 shows this for process reliability with a MTBF target driven by historical data. As can be seen, the desired outcome depends on several different functions. Performing this for each desired outcome, as opposed to looking at system functions in isolation, ensures appropriate consideration of all elements of the process. A functional failure of the vial crimper referenced in Section II impacts reliability targets, for example, while a less detectable fault such is inadequate sealing affects product release and repeatability targets. The division of the criticality in this way means that equipment failure Risk Profile Numbers (RPN) impacting GMP repeatability items do not end up obscured by the higher RPN of equipment impacting reliability and vice versa.

VII. Conclusion

The production of radiopharmaceuticals is a unique and complex regulatory environment. Conventional maintenance methods used in the nuclear sector do not allow for the particular challenges of maintaining a Good Manufacturing Practice facility and conversely those methods used by traditional pharmaceutical companies do not offer adequate solutions for the particulars of nuclear safety.

Capturing all routes of failure is critical particularly during a program of increased production capacity comes into focus. Such scenarios drastically reduce the available downtime for maintenance and force reviews into the efficiency of maintenance activities. Failure to do so leads to unacceptable unscheduled periods of downtime and significant lost revenue.

Consideration of traditional methods of risk based process design ensures maintenance strategies allow for the requirements of both industries. In particular, the outcome-based method of Quality by Design used in pharmaceutical development can assist by showing the cross-functional dependencies of process outcomes and looking at the impact of equipment failures on these outcomes.

Looking forward, review of the applied methods in conjunction with the annual shutdown review offers opportunities for continuous improvement and refinement of the described methodologies into a practical and iterative maintenance system. Key to this is the continued trending and consideration of unscheduled maintenance calls as well as quality assurance out of specification results and batch failure to release statistics to ensure that the applied strategies continue to have the desired results.

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Introduction of an Isotope Production System to CANDU Reactors

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I. Introduction

For many years, radioisotopes have been effectively used in the healthcare industry for sterilization, imaging, and therapy. Recently, the adoption of radioisotopes for targeted treatment of cancers has shown significant promise within the pharmaceutical industry. Specifically, the radioisotope Lutetium-177, Lu-177, has been extensively investigated and documented in the treatment of neuroendocrine and prostate cancers. Studies and clinical trials have shown that Lu-177 can produce long-term remission in patients with metastatic prostate cancer [1].

Canada is a world leader in the safe and stable supply of medical isotopes. The Nuclear Research Experimental (NRX) reactor at Chalk River Laboratories began operation in 1947 and was a pioneer in radioisotope production [2]. The McMaster Nuclear Reactor at McMaster University is the sole supplier of Iodine-125 [3]. With increasing demand for nuclear medicine and quickly evolving medical technology, Canada is leveraging decades of experience and the existing fleet of CANada Deuterium Uranium (CANDU) nuclear reactors to continue as a leader in this expanding industry.

Bruce Power operates eight CANDU reactors between two stations. Currently, the four units in the Bruce B station produce low specific activity (LSA) and high specific activity (HSA) Cobalt-60, Co-60, which is used worldwide for the sterilization of medical equipment and gamma knife treatment of tumours [4]. Bruce Power has now partnered with IsoGen to produce Lu-177 using their nuclear fleet and expand their contribution to the global medical industry [5]. IsoGen is a new company formed by Kinectrics and Framatome, leveraging expertise in safety analysis and design as well as tooling and procurement, respectively.

II. Problem Statement

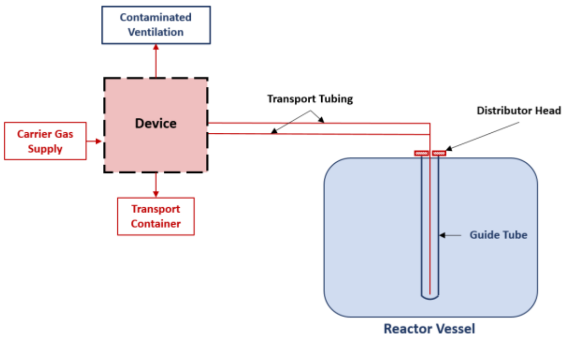
The production of radioisotopes relies on an irradiation source, which has traditionally been through nuclear research reactors. Due to production constraints and limitations of the short isotope half-life, the growing demand is vastly greater than supply for Lu-177. Consequently, the treatment is not always readily available and comes at a significant cost. A stable long-term commercial production source is required to supply Lu-177 to cancer patients worldwide. Bruce Power has demonstrated its ability to produce medical isotopes at Bruce B with the production of Co-60. There is an opportunity for Bruce Power to pursue production of other medical isotopes using their reactors to meet industry demand.

III. Project Scope

An Isotope Production System (IPS) integrated within the Bruce Power CANDU reactors will allow Canada to continue supporting the growing medical isotope market. The purpose of the IPS is to provide a safe and controlled method of producing Lu-177 through neutron capture within the CANDU reactor vessel. The IPS must be designed to operate with minimal impact on power generation and station operations. Successful implementation of the IPS is dependent on three key steps: establishing feasibility, evaluating design considerations and testing, and regulatory approval.

IV. System Description

The IPS will interface with an existing CANDU reactor vessel penetration. Lu-177 is generated through the irradiation of Ytterbium-176, Yb-176. The system will use an inert carrier gas to pneumatically insert Ytterbium Oxide, Yb_2O_3, targets into the reactor then retrieve them after the required irradiation period. A skid outside of containment will be the main operator interface. It will be used to add and remove targets and control system operations. The skid, the gas supply and the reactor penetration will be connected by tubing. A general sketch of the system layout is shown below:



The IPS consists of the following components:

Target: The target material is Yb_2O_3 in powder form which is contained within a completely sealed capsule. The capsule must tolerate the in-core environment of high temperatures and radiation. The number of targets in the system will depend on market demand and production requirements

Reactor Vessel: An existing spare reactor penetration will be used to hold the targets during the irradiation period. A guide tube will be used as pressure boundary to prevent a direct interface between the IPS and the heavy water (D_2O) moderator. A distributor head at the top of the penetration will allow the targets and carrier gas into and out of the guide tube

Delivery Device: The insertion and retrieval of the targets will be operated via a skid located outside containment and away from the reactor vessel. The delivery device houses all the components required to allow an operator to safely control the IPS pneumatics and monitor the location of the targets. The delivery device will supply the pneumatic energy through a carrier gas supply; once spent, the effluent gas is vented.

Transport Container: The irradiated targets are deposited into a transport container. This portable, shielded container is designed to handle the dose and heat produced by the targets. It will protect the targets from damage and allow workers to safely transport them for processing offsite.

In summary, the IPS will facilitate the following basic production cycle:

1. Insert fresh targets into the delivery device and pneumatically convey it into the reactor vessel through tubing.
2. Irradiate the targets for the required period to produce sufficient Lu-177 activity.
3. Pneumatically convey the irradiated targets back to the delivery device ensuring all targets have been retrieved.
4. Deposit irradiated targets into transport containers.
5. Transport offsite for downstream processing and manufacturing

V. FEASIBILITY

Operating experience exists for similar systems deployed around the world. The general premise of a pneumatic skid-based isotope production system is based on Framatome's Aeroball Measurement System (AMS) deployed in over 22 Framatome-KWU reactors. The AMS uses small ball detectors to map neutron flux across the reactor core. An inert gas is used to push the balls into various core locations. The balls are then retrieved after the irradiation period and measured. Similarly, a pneumatic system for various isotope production, including Lu-177, is installed and in operation in the Research Neutron Source Heinz Maier-Leibnitz (FRM II) reactor in Munich. Targets are pneumatically conveyed into channels that penetrate the moderator tank where they are held for the required irradiation period then extracted. Both these systems are controlled from a remote skid located outside the reactor core. The successful operation of these systems abroad and lessons learned will feed into the design and implementation of an IPS in CANDU stations.

Within the Bruce Power CANDU reactors, there are spare penetrations into the reactor vessel. These were originally intended for monitoring or maintenance but now can be repurposed to send targets into the core. Since the target material is Yb_2O_3 powder, the target can be configured into a geometry with a small diameter such that it can fit within the diameter of the existing penetrations. The CANDU reactors also offer a large vertical height within the vessel where there is favourable thermal neutron flux. This can sustain the nuclear reaction rate required to reach the desired Lu-177 specific activities within a relatively short irradiation period.

The target diameter is very small relative to the reactor vessel. The overall impact on the neutron flux and temperature is expected to be negligible. No added burden on the Reactor Regulating Systems (RRS) or fuelling is expected. However, these assumptions must be validated through detailed analysis.

A unique challenge posed by Lu-177 production is the nature of the isotope. Its half-life is approximately 7 days, during which the irradiated targets must be packaged, transported off-site, processed, manufactured into a biomolecule and administered to the patient. Although Bruce Power is situated in a remote location, modern road infrastructure will provide efficient transportation to the nearest international airport. The partnered isotope processors have experience with specialized air transport of nuclear materials and will be able to streamline the downstream logistics.

Beyond just the technical constraints of installing a new system in a nuclear power plant, profitability is also a major consideration for feasibility. There are significant costs associated with design, analysis, materials and installation. Continued isotope production is accompanied by operating costs. However, the market potential of Lu-177 is anticipated to allow for profitable production on a commercial basis without the need for any government subsidies. For Lu-Dotatate, a biomolecule used to treat pancreatic and neuroendocrine cancers, a supply chain already exists in 40 countries [6]. PSMA-617, a Lu-177 labelled therapeutic agent for prostate cancer, is considered the "billion-dollar molecule" due to its potential for treating advanced metastatic cancer [7]. Regulatory approval for its therapeutic use is expected within the year.

Uninterrupted supply of Lu-177 is vital for a reliable cancer treatment for patients worldwide. Bruce Power is the largest operating nuclear site in the world. There are two stations that contain 4 reactors each. If the IPS is successfully implemented in one unit, it can be expanded to other units in order to consistently meet the anticipated growing demand for Lu-177. By using units in both stations, Bruce Power can sustain isotope production even during station wide shutdowns for planned maintenance. In addition, there is potential for the IPS to be adapted to produce other radioisotopes.

Although the IPS will be a prototypical system for a CANDU reactor, Bruce Power has considerable experience producing medical isotopes. The Bruce B station currently produces Cobalt-60. The introduction of the IPS and Lu-177 will add to Bruce Power's existing expertise and strengthen the current partnership with the medical industry.

VI. Design Considerations & Testing

The IPS is a first-of-a-kind design for a CANDU reactor and must undergo various testing prior to being qualified for installation and commissioning. The testing must validate IPS design, material selection and performance.

Throughout the IPS, varying amounts of shielding are required to ensure personnel safety and satisfy regulatory requirements. The design and operation of the IPS will comply with CNSC REGDOC-2.7.1, Radiation Protection, which outlines the requirements and guidance on the topics of radiation protection programs, worker dose control and radiological hazard control [8]. Shielding will be required for the delivery device, transport container and the distributor head at the reactor vessel penetration. Since operators will control the flow of the targets through the IPS from the skid, adequate shielding is required to allow them to work closely with the control and pneumatic panels. Targets will be shipped using the transportation container and must meet packaging regulations for the transportation of radioactive materials as per IAEA Regulations for the Safe Transport of Radioactive Material [9]. Finally, shielding at the distributor head will prevent a possible neutron radiation beam produced by the reactor to be released outside of containment. Potential candidates for shielding materials range from depleted uranium and borated polyethylene to more conventional alternatives including lead and tungsten. Shielding efficacy will be validated through a combination of safety analysis and testing.

The IPS system will interface with other systems and must minimize any adverse impact to their system functions. For example, the pneumatic flow of the IPS will be controlled by a carrier gas. This gas will be in contact with the irradiated targets and will need to be directed to the contaminated ventilation system when discharged from the IPS. Additionally, lifting and craning capabilities to support hauling of the transport container will be considered in the design. The container lid will be adapted to allow for appropriate rigging protocols based on the anticipated weight.

Each major component of the IPS will be mocked up to determine the optimal routing path of the tubing and select the preferred location and footprint of the delivery device and transport container. The mock-ups will reflect the conditions and layout in the station so that the IPS operation can be accurately simulated. This includes carrier gas pressure and temperature, target velocity, tubing height and distance, and transfer of targets into the transport container. During this evaluation of the IPS design and function, human factor considerations will be applied from an operability and maintainability perspective. This will focus on operator interface with the IPS controls, access and position of components and potential human error precursors. Mock-ups of the IPS will assist in identifying potential component and system failures and enacting appropriate mitigating measures. Failure modes can be assessed by testing the mock-ups at extreme conditions.

Lastly, formal testing of the major components described above will qualify the equipment to the applicable standards and regulations. Parameters such as leak tightness of the target, tubing and transport container will be evaluated. The target strength and durability will be assessed to ensure that its integrity is not compromised during operation since it will travel at high speeds through the IPS tubing. The system tubing will be subjected to a pressure test. The mock-up will also be used to validate the optimal tubing bend radius for smooth passage target. Since the IPS can operate with various number of targets, mock-up testing will include all bounding scenarios.

VII. Regulatory Approval

The IPS introduces a new use for the CANDU nuclear station. In addition, this system interfaces with the reactor vessel, exposes materials to neutron flux and yields radioactive products. As a result, regulatory approval is required upon a thorough review of the IPS design and analysis.

A modification or a new system that is added to a Canadian nuclear power plant must be classified per CSA N285.0 [10]. Classification establishes the design basis for the system and its components considering the system's importance to nuclear safety based on the consequences of its failure. The IPS will be designed to not interfere with any nuclear safety functions; however, the system does penetrate the reactor vessel through an existing containment boundary. As such, portions of the IPS are identified as containment boundary extensions, Class 2, and the remaining as non-nuclear class, Class 6. The proposed class boundaries must be submitted to the CNSC in a letter for approval.

The majority of the IPS will experience the pneumatic pressure used to convey targets from the skid to the reactor vessel and back. The IPS will be a pressure boundary system per CSA N285.0 and will require registration with the provincial administrative authority; in Ontario, this is the Technical Standards and Safety Authority (TSSA) [11]. Registration requires support documentation to be provided. For Class 2, this includes CNSC approved classification list, system flowsheet, design specifications, stress analysis report, and overpressure protection report.

Currently, the Bruce Power's Power Reactor Operating License (PROL) does not include Lu-177 production. A license amendment request and approval from the CNSC is required prior to installation and operation of the IPS. To facilitate an amendment, Bruce Power must prepare a safety case which demonstrates that the IPS does not introduce any unmitigated nuclear risks. The safety case is built on analysis of the IPS impact on the station, particularly the safety systems functions,

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and must comply with CNSC REGDOC 2.4.1 and 2.4.2 [12][13].

The safety case will include assessments of accident scenarios. Deterministic analysis will be used to evaluate the impact of the IPS on the propagation of existing design basis accidents, primarily inside the reactor vessel. Probabilistic analysis will use the failure modes of the IPS to identify and classify new initiating events for accidents. The bulk of the analysis, however, will assess the impact of the IPS installation and operation on reactor operation. This includes:

Core physics: Adding new materials (targets, carriers, tubing, etc.) into the core will absorb neutrons. What is the impact on the local neutron flux? Will this require adjustments by the Reactor Regulating Systems (RRS)?

Thermal hydraulics: Irradiation of targets will generate heat. Are heat transfer properties of in-core materials sufficient to dissipate the heat into the moderator? Is there any significant local moderator heating?

Fuel management: Negative reactivity in the space occupied by the IPS may have some local effects on fuel channels. Does this introduce any significant operation changes to fuelling?

Safety Systems: Irradiated materials will move through the reactor vessel during target retrieval. What is the impact of the moving radiation source on in-core detectors? Can this result in an operational event such as an alarm, setback, or reactor trip?

Canada is a signatory to the Treaty on Non-Proliferation of Nuclear Weapons which is overseen by the International Atomic Energy Agency (IAEA) [14]. The IAEA will be engaged throughout the design process as a stakeholder; based on their input, safeguards such as monitoring and restricted access may be implemented.

VIII. CONCLUSION

The three focus areas described in this paper are key steps to ensure the IPS is safely installed and operated in a CANDU reactor. Feasibility was established based on first validating that the neutron flux in a CANDU reactor vessel is sufficient to achieve sufficient Lu-177 activation and then identifying options for the system to access the reactor core. There is significant operating experience of similar systems worldwide and the partnered organization provide a breadth of strong technical proficiency in the various facets of the project. The existing demand and medical potential of Lu-177 demonstrate the commercial viability of the project.

As the project moves into detailed design, other design considerations such as system interfaces and human factors will be incorporated. Mock-up the major components of the IPS as well as the full system will effectively run through anticipated production cycles under both normal and abnormal system conditions. Shielding is an important feature for several components in the IPS and will be evaluated through deterministic safety analysis and testing.

Regulatory approval of the IPS through a license amendment is contingent on thorough analysis of the impact on the reactor and nuclear safety. The system effects on reactor operating conditions are anticipated to be negligible due to the small relative size of the in-core IPS components; however, this must be validated through deterministic analysis. Probabilistic analysis will also be employed to identify potential new accident scenarios and apply mitigating measures.

The IPS will provide a safe, controlled, and sustainable source of Lu-177. Once proven safe and reliable in a CANDU reactor, this system can be expanded and adapted to produce many other radioisotopes with a wide array of medical uses. Successful implementation of this project has the potential to save lives at home and abroad. It draws on Canadian nuclear expertise for an innovative design and safety analysis and will keep Canada a leader in medical isotope production.

ACKNOWLEDGMENT

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Developing and maintaining internationally traceable radionuclide standards to benefit Australian nuclear medicine and industry

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I. INTRODUCTION

Radionuclide metrology underpins radioactivity measurements in the $9.6 billion global radioisotope market, which includes over 40 million nuclear medicine procedures performed annually [1,2]. Radionuclide metrology is the study and measurement of radionuclides, with the aim to develop accurate measurements of radionuclide activity and disseminate this in the form of standards. Standards are measured using primary methods utilising principles that do not require reference to a calibration [3]. To provide assurance of the result's accuracy, international comparisons of activity measurements are performed between collaborating radionuclide metrology laboratories around the world.

This international approach gives users in a wide range of nuclear industry, assurance of accurate and consistent results when taking measurements or calibrating instruments with reference to these radionuclide standards. A real world example can be given in the field of nuclear medicine. Calibration to radionuclide standards ensures patient doses are traceable and accurate within uncertainties to benefit patient treatment.

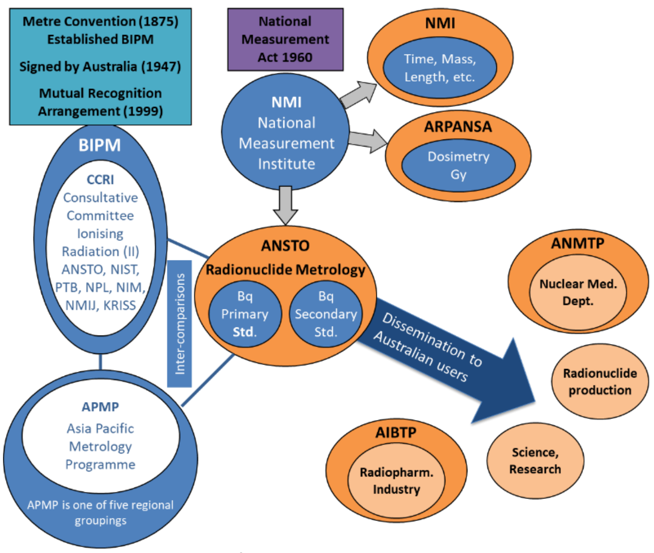

Figure 1. Summary of International and Australian legal metrology arrangements.

International and Australian legal metrology arrangements are summarised in Fig. 1. This shows where Australia and ANSTO fits into the overall structure. International standards were introduced with the Metre Convention (1875) which established the Bureau International des Poids et Mesures (International Bureau of Weights and Measures) (BIPM). Member states coordinate through the BIPM on matters pertaining to measurement science and standards [4]. Under the BIPM sits various committees and regional groups contributing expertise to measurement science from around the world. Consultative Committees (CCs) are scientific bodies created to assist the BIPM in its scientific work. The Consultative Committee for Ionizing Radiation, Section II: (CCRI(II)), of which ANSTO is a full member, focuses on the measurement of radionuclides.

The Comité International des Poids et Mesures (International Committee of Weights and Measures) (CIPM) aims to promote uniformity in measurement units through the Mutual Recognition Arrangement (MRA) [5]. The CIPM MRA provides a formal framework through which National Metrology Institutes demonstrate the equivalence of their measurement standards and the certificates they issue, through intercomparisons and quality systems. The outcomes are internationally peer-reviewed Calibration and Measurement Capabilities (CMCs) [5] through which one State's standards can be accepted by other signatories of the CIPM MRA.

The Australian National Measurement Act 1960 appoints the Chief Metrologist and designates the National Measurement Institute (NMI) as the peak body responsible for maintaining Australia's measurement system. ANSTO, as the centre of nuclear science and technology in Australia, is appointed as the institute responsible for maintaining the Australian standard for radioactivity. Under authorisation from the Chief Metrologist, Radionuclide Metrology at ANSTO works to develop, maintain and disseminate the Australian standard for radioactivity.

Australia's program to develop, maintain and disseminate the standard for radioactivity are described with regard to a recent international standardisation campaign.

II. DEVELOPING AUSTRALIA'S PRIMARY STANDARDS

Primary standards are developed by applying specialised measurement techniques that determine the activity of a radionuclide source that determine the activity of a radionuclide source measured using physical principles, i.e. without reference to prior calibration. The complex and unique decay scheme of each radionuclide will influence the technique through which the primary standard is achieved.

The CCRI(II) recognized the industry need for an internationally verified Ge-68 standard to support increasing application of Ge-68 and its daughter radionuclide, Ga-68, for nuclear medicine applications. In 2015 the National Institute of Standards and Technology (NIST, USA) piloted an international comparison and distributed aliquots of the same Ge-68 solution to 17 individual participants. Participants standardised the solution applying their best possible methods and submitted their activity concentration values to the organiser [7].

ANSTO performed the standardisation of Ge-68 using the 4πβ-γ coincidence extrapolation method [6]. This technique can be applied to radionuclides that decay by the simultaneous emission of two different types of radiation. A custom-built detection system that contains two channels, each sensitive to only one type of radiation, is used for measurement (Fig. 2). Analysing the data for instances where both channels detected a disintegration at the same time provides information on the detection efficiency, and ultimately the source activity. For this work, liquid scintillation counting was applied for the detection of positrons and a NaI(Tl) detector for counting annihilation photons from Ga-68 (in radioactive equilibrium with Ge-68) [6].

Using the 4πβ-γ coincidence extrapolation method ANSTO's result was reported as 620.2 ± 3.4 kBq/g. The Comparison Reference Value was calculated as 621.7 ±1.1 kBq showing that ANSTO's method provided a precise result in excellent agreement with many other international facilities (Fig. 3) [7]. The full results of the intercomparison are published by Cessna et al. 2018 [7].

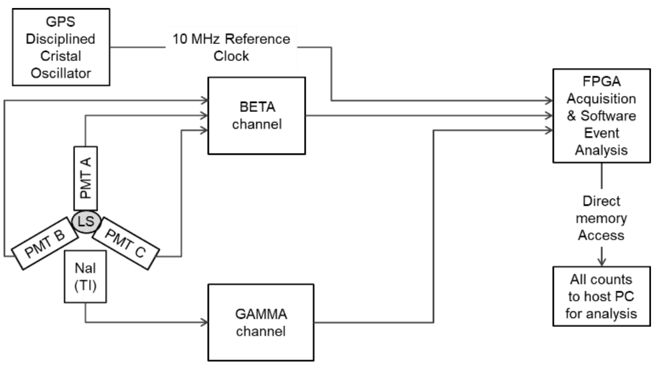

Figure 2. Schematic of the 4πβ-γ coincidence extrapolation equipment.

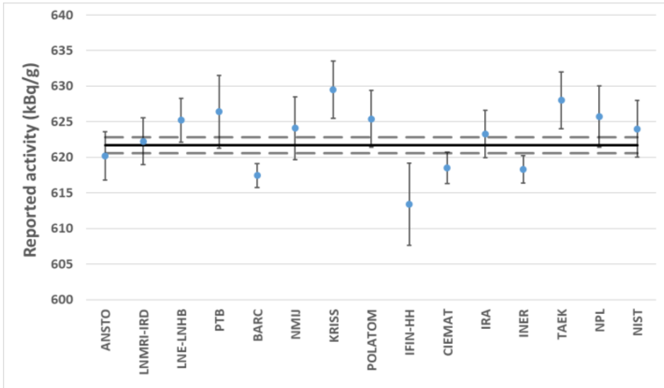

Figure 3. Results of 17 laboratories (outliers removed) for measurements of a Ge-68/Ga-68 solution as collated by CCRI(II) [7].

Once the primary standard value is obtained, it is maintained at ANSTO by transferring it to the Secondary Standard Ionisation Chamber (SSIC) by developing a radionuclide- and geometry-dependent calibration factor from the standardised solution. This factor is able to be maintained in the SSIC long after the more transient radioisotopes have decayed and can be used to assign precise radioactivity values to future samples.

III. GE-68/GA-68 DISSEMINATION TO NUCLEAR MEDICINE

The radionuclide Ge-68 decays purely by electron capture, giving rise to the emission of low energy X-rays and Auger electrons [6]. It has a relatively long half-life (270.95 (26) days), decaying to Ga-68 which has a short half-life (67.83 (20) minutes) and decays predominantly by positron emission to Zn-68 (stable) [8].

The daughter radionuclide, Ga-68, is increasingly being used in quantitative Positron Emission Tomography/X-ray Computed Tomography (PET/CT) [6]. The parent radionuclide, Ge-68, is used to manufacture generators that can provide a steady supply of Ga-68 for a long period of time, without the need for an onsite cyclotron [6]. A Ge-68/Ga-68 solution in equilibrium is a convenient surrogate for calibration of nuclear medicine devices for Ga-68, since the low energy emissions of Ge-68 are undetectable by ion chambers allowing for no interference in measurements.

After participation in the CCRI(II) intercomparison, an internationally verified and traceable calibration factor for Ge-68/Ga-68 was available at ANSTO through the SSIC (Fig. 4). The standard geometry for the SSIC is 3.6 mL of liquid in a sealed glass ampoule. A new solution of Ge-68/Ga-68 was purchased and the activity concentration determined by dispensing an aliquot into this standard geometry and measuring in the SSIC (Fig. 4). This solution was then dispensed in triplicate into the measurement geometry used for measuring patient doses in nuclear medicine (8 mL in 25 mL vials) and these vials measured in the SSIC. As the activity concentration

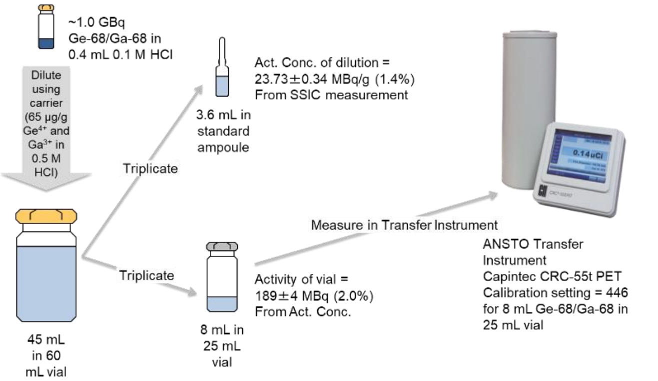

Figure 4. Transfering standard measurements to industry geometries and the ANSTO Transfer Instrument.

of the solution was known, the impact of the new vial geometry could be quantified and a new SSIC calibration factor determined. This solution, linked to the primary standard through the SSIC, was also measured on a Capintec CRC-55tPET ion chamber which was established by ANSTO as a transfer instrument for the dissemination of short-lived radionuclide standards. The transfer instrument can be transported to clients' sites to measure and compare radionuclides for traceability to the primary standard. This dissemination method is used for measurements where a short half-life prevents the source being transported back to ANSTO for direct SSIC measurements.

ANSTO provided measurement traceability to a nuclear medicine facility for Ga-68 in the vial and syringe geometries applied for clinical work. The Ge-68/Ga-68 vial certified by measurement in the SSIC, together with the ANSTO transfer instrument, were taken to the nuclear medicine laboratory. A Ga-68 solution eluted from the nuclear medicine facility generator was standardized by dispensing an accurately weighed amount of solution into a vial and measuring the resulting source in the ANSTO transfer instrument. Both vials were also measured in the nuclear medicine facility ion chamber to assess the accuracy of their default calibration setting. An accurately weighed mass of the standardized Ga-68 solution was drawn up into a syringe and measured in the nuclear medicine ion chambers to determine the effect of geometry variation on measurement. The experiment design and results are summarized in Fig. 5 and Table 1.

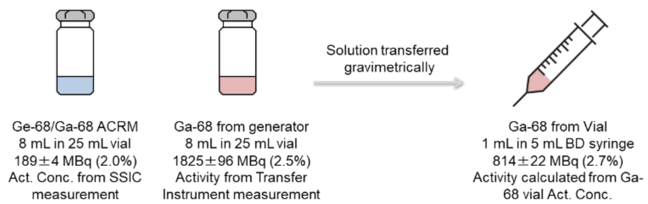

Figure 5. Schematic of the acquisition of the Ge-68/Ga-68 and Ga-68 geometries.

TABLE I. GEOMETRY EFFECTS OF GE-68/GA-68 MEASUREMENTS

Geometry	Ref. Activity (MBq)	ANSTO Transfer Instrument Capintec CRC-55t PET			Capintec PET dose calibrator		
		Factor	*Result (MBq)*	*Deviation (%)*	*Factor*	*Result (MBq)*	*Deviation (%)*
Ge-68/Ga-68 – vial	3825 (96)	446	189	NA	460	188	-0.67
Ga-68 – vial	3825 (96)	446	3825	NA	460	3810	-0.41
Ga-68 - Syringe	814 (22)	-	-	-	460	850	4.4

The results show that when the vial factor is used to measure a syringe geometry, the activity can be overestimated by more than 4%. Geometry effects are particularly noticeable with Ga-68 due to the high energy of beta emissions that can interact with the source encapsulation and the detector, producing Bremsstrahlung radiation and skewing radioactivity measurements [6]. This is of particular importance when administering patient doses to ensure positive health outcomes through accurate dose administration.

IV. DISTRIBUTING STANDARDS TO STAKEHOLDERS

ANSTO currently has two programs through which Australian standards are made available to national stakeholders. The Australian Industry Becquerel Traceability Program (AIBTP) focuses on the industrial production of radioisotopes, specifically cyclotrons for F-18 and for radiopharmaceuticals produced on site at ANSTO. The AIBTP offers measurement assessment and calibration (if required) by a radionuclide metrologist, as well as analysis of linearity and stability data, to deliver quality assurance of measurement accuracy to these stakeholders.

Many of the stakeholders of AIBTP are interested in F-18 which has a specific challenge of a relatively short half-life (1.82890 (23) hours) [9]. Sources with high enough radioactivity to enable measurement comparison between the client ion chamber and the SSIC to certify the source activity would cause safety concerns regarding dose to personnel. The Capintec PET Transfer Instrument detailed in Section III which has been closely studied for stability, is linked to the SSIC. Factors from the SSIC are replicated in the transfer instrument so this can be sent to clients to measure manufactured F-18 with traceability to the SSIC secondary standard (and therefore the Australia primary standard) at their site. This reduces the overall radioactivity required to perform the assessment as transportation of the radionuclide is not necessary. This transfer instrument means that access to short lived F-18 standards is available to not just the closest hospitals geographically to the SSIC but can be taken all over Australia without a compromise in safety.

The Australian Nuclear Medicine Traceability Program (ANMTP) developed in collaboration with the Australian and New Zealand Society of Nuclear Medicine (ANZSNM) is run annually from September-December. Each year, nuclear medicine departments and clinics across Australia are invited to participate in the program. A radionuclide metrologist from

ANSTO visits the client's site to perform a measurement assessment on their ion chambers using the radionuclides that are offered in that year. The standards offered depend on client demand, the availability of primary standards and radioactive source availability. If required, adjustment of ion chamber factors are performed to bring the measurements closer to the standardised value.

ANMTP is run at cost recovery and consistent pricing is offered to every hospital regardless of location to ensure equal access to all nuclear medicine departments across Australia. Running a coordinated program with multiple participants annually allows the cost of sources to be split amongst participants, reducing the overall costs of disseminating radioactivity standards.

To date ANMTP has been running for 5 years visiting 27 different Australian hospitals across 6 states. 5 different ACRMs have been tested on 57 individual ion chambers with a total of 296 disseminations of the Australian standard being performed. Ge-68/Ga-68 was offered to ANMTP subscribers in 2017-2018 making use of the internationally verified primary standard obtained through the CCRI (II) comparison. Round 6 of ANMTP is set to commence in November 2019 and the standard for Lu-177 will be included in the program for the first time.

V. Conclusion

The global radioisotope industry is underpinned by radionuclide metrology, recognised internationally, and supported by ANSTO in Australia. Radioisotopes of importance to industry are identified and standardised by primary methods involving first principle methods. Results are compared with those from other international radionuclide metrology laboratories to establish the equivalence of different international standards.

Once developed, primary standards are maintained in Australia using the SSIC for the geometries used by stakeholders. The standards are disseminated to stakeholders to provide traceability to the Australian standard, which is directly linked to international standards. Future priorities include extending the portfolio of Australian standards and contributing to the international community by participating in further intercomparisons. These activities will focus on radionuclides with emerging medical or industrial significance. Through this, the offered standards for the ANSTO run programs will also have the potential to expand.

Traceability to internationally verified primary standards improves the accuracy of radioactivity measurements, bring confidence of best practice and ultimately, benefits Australian industry and recipients of nuclear medicine.

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Study of environmental samples utilizing the neutron activation analysis at the training reactor VR-1

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I. INTRODUCTION

This research work deals with the study of environmental samples, particularly alluvial soils and mosses, using the instrumental neutron activation analysis (INAA) at the VR-1 training reactor. Neutron activation analysis (NAA) together with the gamma-spectrometry are often utilized to determine the quantitative and qualitative composition of the sample.

I. A. Why study our environment?

It is important to know the elementary composition of the environment (i.e. in air, water and soil) and the presence of contaminants for the assessment of environmental pollution. The attention is paid especially to heavy metals, such as cadmium, mercury or lead due to their toxicity. Nowadays, the living organisms as indicators of environmental pollution are increasingly used. Mosses and lichens are most often used organisms, which indicate the presence of pollutants in the environment. Soils and mushrooms can also be analysed. In this paper, alluvial soils and mosses are investigated.

As said before, neutron activation analysis can be used for the composition determination of the environmental samples. Using the NAA, heavy elements such as mercury, arsenic or lead can be measured. Also, metals like sodium, manganese, magnesium or aluminum can be analyzed. The huge advantage of this method is the possibility of gaining 30 – 40 elements from a single experiment. [2]

Environmental samples were also studied in other countries. For example, some kind of mosses from Opole Voivodeship in Poland or alluvial soils from the Siwa Oasis in Egypt were studied at the Russian IBR-2 reactor [3, 4]. Similarly, the soil and plant samples from around the Manzala lake in Egypt, where a high pollution was expected, were irradiated at the ET-RR-1 research reactor [5]. Also, the soil samples from the cities of Zacatecas and Guadalupe in the state of Zacatecas, México due to the mining industry and rapid growth of vehicle traffic were analyzed, and the experiment took place at the University of Missouri Research Centre at the local research reactor. The amount of arsenic, barium, zinc, iron or manganese in the samples were higher than the amount in the samples marked as heavily polluted (according to US EPA legislation) [6].

II. MATERIALS AND METHODS

II. A. Training reactor VR-1

The VR-1 reactor is a pool-type light water reactor with a maximum power of 80 W. It is operated by the Faculty of Nuclear Sciences and Physical Engineering at the Czech Technical University in Prague. The reactor is commonly used for education or for experiments. Currently, the reactor uses IRT-4M fuel with enrichment of 19.7 % of ^{235}U. As a moderator, coolant and reflector, the demineralized light water is used. Due to the low reactor power, it is not necessary to use pumps to dissipate heat. The reactor core is consisted of 15 – 20 fuel assemblies and 5 – 7 cadmium control rods. For experimental purposes, dry vertical and horizontal channels are situated in the reactor and they can be utilized for irradiation of samples and neutron activation analysis research tasks. [2, 7]

II. B. Neutron activation analysis

Neutron activation analysis is a method utilizing the nuclear reactions and radiation emitted by activation products to determine the composition of the samples. The nuclear reactions are induced by neutron bombardment of the material. Gamma radiation has an important role in neutron activation analysis. It is measured after irradiation of the sample.

Main advantage of the NAA method is the possibility to perform the non-destructive way of analysis (instrumental NAA) [8] and capability of measuring of trace elements. [9]

The instrumental neutron activation analysis consists of the following steps. Before inserting a sample into the reactor core, it is necessary to dry it (e.g. in the oven), weigh and encapsulate it (see Fig. 1). Then the sample is loaded into the neutron field of the nuclear reactor. After that, the sample is irradiated for requested time period. Subsequently, it is removed from the neutron field and transported to the γ-ray laboratory, where the gamma spectra of the sample are measured. Eventually the composition of the sample is identified.

Figure 1. The prepared and encapsulated samples of alluvial soil.

During the irradiation, the activity of the sample increases. The main output of γ-ray spectrometry is the production rate of neutrons. It is expressed by the equation [1]:

$$P = \frac{S_\gamma \lambda \frac{t_{real}}{t_{live}}}{I_\gamma \varepsilon_\gamma \left(1-e^{-\lambda t_a}\right) e^{-\lambda t_v}\left(1-e^{-\lambda t_{real}}\right)}, \quad (1)$$

where S_γ is the area under the full energy peak, λ the decay constant of radionuclides, t_{real} the real time of measurement, t_{live} the live time of measurement after subtraction of the dead time, I_γ the gamma line intensity, ε_γ the absolute detection efficiency for E_γ energy, t_a and t_v the irradiation and cooling time. Numerical value of the production rate equals to saturated activity A_{sat}. [1]

To determine an elemental composition in the sample, the relative method is used. The sample and the referenced standard (i.e. object made of the same element, which is possibly contained in the sample) is irradiated under the same conditions (the same irradiation time and neutron field). Mass of the analyzed element, which occurs in the specimen, is obtained from the equation [8]:

$$m_{element} = \frac{m_{standard} A_{element}}{A_{standard}}, \quad (2)$$

where the activity A is equaled to P.

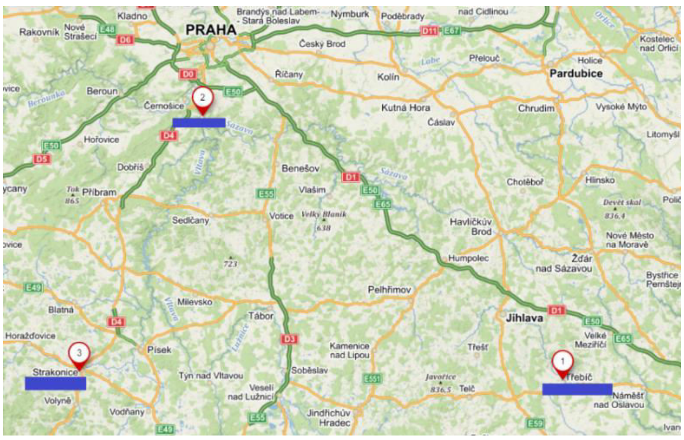

Figure 2. Places, where the samples were taken. [10]

II. C. Environmental samples

Three samples of alluvial soils from three places in Czech Rebublic were obtained - three samples from the bank of the Jihlava river close to the town Třebíč, then the same amount of samples from the banks of Vltava river nearby Prague and Volyňka river not far from Strakonice town. Two types of mosses were also collected, particularly two samples of Pleurozium schreberi closed by Třebíč and Hypnum cupressiforme near Praque. Places, where the samples were collected, are displayed in Fig. 2. These locations were chosen based on the longer distances from each other and easy accessibility. Moreover, nearby Volyňka river lies company producing components for automotive industry or the aluminium castings.

III. Results

Nine samples of soils and three samples of mosses in total were irradiated in three experiments of the NAA. Before each experiment, the samples were collected and dried in the oven at 105 degrees Celsius for 24 hours. Subsequently, they were encapsulated and weighed. Investigated environmental samples together with activation standards were attached to the plastic rod and put into the neutron field. In the reactor core, the samples were irradiated for one hour. After removal from the reactor, the gamma lines emitted by the samples were measured using the semiconductor HPGe detector. The semiconductor HPGe detector is shown in Fig. 3 and Fig. 4. Based on the obtained experimental data, the composition of samples was determined. Elements included in all samples and their nuclear properties are summarized in Table I. The toxic elements observed in the samples of alluvial soils are presented in Tables II, III and IV. The qualitative analysis of mosses was also performed. Elements different from those found in soils are given in Table V. The quantity of some elements was determined as well, namely titanium in soil samples from Jihlava and Vltava river and aluminum in soil samples from Volyňka river. Results of quantitative NAA are presented in Table VI. Energies of gamma lines and half-lives were obtained from [11].

Figure 3. Inside of semiconductor HPGe detector.

Figure 4. Outside of semiconductor HPGe detector.

TABLE I. ELEMENTS CONTAINED IN ALL SAMPLES

All samples			
Observed radionuclide	***Energy of gamma line (keV)***	***Half-life***	***Nuclear reaction***
^{24}Na	1368.6; 2754.0	14.96 h.	^{23}Na(n,γ)^{24}Na
^{56}Mn	846.7; 1810.8; 2113.1; 2522.8; 2657.5	2.58 h.	^{55}Mn(n,γ)^{56}Mn ^{56}Fe(n,p)^{56}Mn
^{42}K	1524.7	12.36 h.	^{41}K(n,γ)^{42}K
^{51}Ti	320.1	5.76 min.	^{50}Ti(n,γ)^{51}Ti
^{52}V	1434.1	3.74 min.	^{51}V(n,γ)^{52}V
^{28}Al	1779.0	2.24 min.	^{27}Al(n,γ)^{28}Al

TABLE II. TOXIC ELEMENTS IN SOILS IN JIHLAVA RIVER

Jihlava river			
Observed radionuclide	***Energy of gamma line (keV)***	***Half-life***	***Nuclear reaction***
^{139}Ba	165.9	83.06 min.	^{138}Ba(n,γ)^{139}Ba
^{203}Hg	279.2	46.61 d.	^{202}Hg(n,γ)^{203}Hg
^{76}As	559.1	1.08 d.	^{75}As(n,γ)^{76}As

TABLE III. TOXIC ELEMENTS IN SOILS IN VLTAVA RIVER

Vltava river			
Observed radionuclide	***Energy of gamma line (keV)***	***Half-life***	***Nuclear reaction***
^{76}As	559.1	1.08 d.	^{75}As(n,γ)^{76}As
^{139}Ba	165.9	83.06 min.	^{138}Ba(n,γ)^{139}Ba

TABLE IV. TOXIC ELEMENTS IN SOILS IN VOLYŇKA RIVER

Volyňka river			
Observed radionuclide	***Energy of gamma line (keV)***	***Half-life***	***Nuclear reaction***
^{76}As	559.1	1.08 d.	^{75}As(n,γ)^{76}As

Volyňka river			
Observed radionuclide	***Energy of gamma line (keV)***	***Half-life***	***Nuclear reaction***
^{82}Br	776.5; 554.3	35.30 h.	^{81}Br(n,γ)^{82}Br
^{139}Ba	165.9	83.06 min.	^{138}Ba(n,γ)^{139}Ba
^{203}Hg	279.2	46.61 d.	^{202}Hg(n,γ)^{203}Hg

TABLE V. ELEMENTS CONTAINED IN ALL MOSSES

All mosses			
Observed radionuclide	***Energy of gamma line (keV)***	***Half-life***	***Nuclear reaction***
^{38}Cl	2167.4; 1642.7	37.24 min.	^{37}Cl(n,γ)^{38}Cl
^{128}I	442.9	24.99 min.	^{127}I(n,γ)^{128}I
^{41}Ar	1293.6	109.34 min.	^{40}Ar(n,γ)^{41}Ar

TABLE VI. RATE OF ELEMENTS IN SOIL SAMPLES

River	***Titanium rate (%)***	***Uncertainty (%)***	***Aluminium rate (%)***	***Uncertainty (%)***
Jihlava	0.5	36.2	-	-
Voltava	0.2	23.0	-	-
Volyňka	-	-	9.0	3.5

IV. DISCUSSION AND CONCLUSIONS

In three NAA experiments performed at the VR-1 training reactor, nine samples of alluvial soils and three samples of mosses were investigated. Manganese, potassium, titanium, sodium, vanadium and aluminum were found in all samples. Other, often observed elements included e.g. lanthanum and magnesium. Also, toxic metals, such as arsenic, barium or mercury were discovered. The abundance of titanium in soils of Vltava and Jihlava river and the amount of aluminum in soils of Volyňka river were determined as well. The biggest amount of titanium was found in Jihlava river with a value of 0.48%; in Vltava river, it was found about 0.21% of titanium. These two values are comparable to the values of abundance of titanium in the Earth's crust. The amount of aluminum measured in sample from Volyňka river was approximately 9%, that is slightly higher value than the percentage of aluminum in the Earth's crust. Slight increase in amount of Al could be caused by nearby factory, where aluminum alloys are made. The abundance of elements in the Earth's crust is shown in Fig. 5. Elements, which was found in all samples, are also a part of the Earth's crust. Toxic elements found in specimens are mostly gained from the mining industry. They can occur as ores or as a part of other sources, especially fossil fuels. They can also be released to the atmosphere because of natural storylines, such as volcanic eruptions etc.

Except inorganic samples, the samples of mosses were also irradiated, and the qualitative analysis was performed. In these organic samples were found the same elements as in the samples of soils. However, iodine, argon or chlorine was also observed. It is due to the fact, that the soils contain higher concentration of sodium and manganese. Their gamma lines are very intensive and overlap the gamma lines of the other elements with similar energies. Mosses do not have such high amount of these elements, so in the long-term irradiation it is possible to measure elements with similar energies. Argon occurred in mosses as well, because it is included in the atmosphere.

The obtained data can be useful for assessment of presence of selected problematic elements in environment. It can be said that e.g. mushrooms, lichen, mosses or groundwater in the investigated geographical area can be also analyzed, and the results can be compared with the results of this experiment (as can be seen above).

Neutron activation analysis of alluvial soils and mosses was successfully performed at the training reactor VR-1, so it can be said that the VR-1 reactor is suitable tool to identify the composition of the environmental samples. It can provide useful data utilizable for the determination of environmental pollution.

In the future, more soil samples and mosses will be analyzed, and concentration of some toxic elements will be examined. Based on results obtained in this work and from subsequent irradiation experiments of samples from other parts of the Czech Republic, the visual map showing the environmental pollution will be finally compiled.

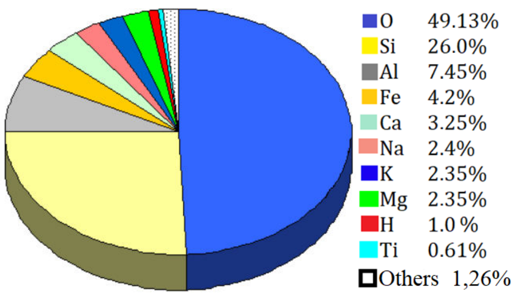

Figure 5. Abundance of elements in Earth's crust. [12]

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Radiosensitivity of *Alocasia maquilingensis* Merr. seeds

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I. Introduction

Alocasia is one of the most diverse and morphologically interesting genera in the family Araceae consisting of medium size to rarely aborescent and gigantic, seasonally dormant to evergreen herbs [1]. Among the cultivated aroids, *Alocasia* is considered one of the favorites of both collectors and landscapers around the world due to their gorgeous leaves that are heart or spade shaped and beautifully colored in gleaming shades of green, purple, bluish green, red, bronze and their combinations [2]. The recent taxonomic review [3] revealed that most of the Philippine *Alocasia* are of ornamental or of potential ornamental value with at least eight (8) species currently being circulated in the international market. Some of these endemic species are in the verge of being vulnerable (*A. micholitziana* Sander; *A. zebrina* Schott van ex Houtte) or endangered (*A. sanderiana* W.Bull) due to over-collection and habitat destruction [4].

One of the interesting and unique Philippine endemics is *A. maquilingensis* Merr., the only species with hairy leaf and petiole (puber group) and characterized by its ivory-white fruiting spathe (Fig.1). Scattered in Luzon, Mindanao, Leyte, and Panay, *A. maquilinensis,* can be found in limestone areas on slopes and primary rain forest at low to medium elevation where it thrives near flowing waters [5]. Although not considered a threatened species yet now, uncontrolled deforestation activities will dry up their habitat and eventually ruin their population. In order to relieve the over-collection from the wild, tissue culture techniques were already established to exponentially propagate rare and endangered species [6, 7] and thereafter reintroduce them to their native habitats. However, with the significant effects of climate change, rebuilding the plant community in their respective habitats will not suffice without the added scientific interventions, like mutation breeding technology.

Mutation breeding is the purposeful application of mutagens (gamma radiation or chemical) in improving plants for creating novel traits such as drought tolerance, disease resistance, breakage of dormancy, improved yield and unique aesthetic values (color, shape or form changes) which may not be originally observed in their wild forms. Compared to recombination breeding (traditional plant breeding or hybridization), mutation breeding has the advantage of correcting a defect in an otherwise elite variety, without losing its agronomic and outstanding traits [8]. Induced mutation has created various elite cultivars registered in the mutant variety database of the IAEA (https://mvd.iaea.org/) ranging from high yielding agricultural crops to the most attractive ornamental plants. This paper is part of an on-going mutation breeding project in Philippine *Alocasia* aiming to develop new varieties for economic and conservation purposes. We first report here the effects of increasing gamma radiation doses in the seed germination of *Alocasia maquilingensis* Merr.

Figure 1. *Alocasia maquilingensis* Merr. with intact infructescence

II. MATERIALS AND METHODS

A. *Plant materials and irradiation treatment*

The immature infructescence of *A. maquilingensis* were collected and air-dried overnight. Still intact, infructescence of *A. maquilingensis* were irradiated with 10, 20, 30, 40, and 50 Gy gamma energy using Gamma Cell 220® ^{60}Co source at the Irradiation Services Section under the Nuclear Services Division of Philippine Nuclear Research Institute (PNRI).

B. *In vitro culture conditions and radiosensitvity study*

Procedures on surface sterilization and aseptic culture were modified from existing literatures [7, 9] using immature seeds as explants. Irradiated seeds and the control (non-irradiated) were washed with soap and diluted fungicide for 15 minutes. Explants were further sanitized using 20% commercial bleach (Zonrox™) for 10 minutes followed by immersion in 85% ethyl alcohol and agitated for 15 minutes. The seed materials were then soaked in 10% H_2O_2 for 15 minutes. Seeds were rinsed 3 times after each stage with sterile distilled water before inoculating on the surface of modified semi-solid Murashige and Skoog (MS) culture medium supplemented with 30 g sucrose. The seeds were incubated under laboratory conditions. Seeds were considered germinated when they exhibited a radical extension of >2mm. Observation of seed germination was made on daily basis for a period of three weeks. Measurement of shoot length and number of roots were done after two months of incubation. Determination of lethal dose 50 (LD_{50}) was done using probit analysis [10].

III. RESULTS AND DISCUSSIONS

A. *Radiosensitivity study of A. maquilingensis* seeds

The radiosensitivity study conducted for *A. maquilingensis* immature seeds revealed a moderate positive relationship between the lethality and doses administered. The LD_{50} of *in vitro* cultures of *A. maquilinguensis* seeds by gamma irradiation in two months after treatment is 46.7 Gy (Fig. 2). Our results are somewhat similar to the radiosensitivity study performed on the seeds of *Anthurium andreanum* with an LD_{50} of 40Gy [11]. On the other hand, the study on *Colocasia esculenta* L. [12] showed that the effective mutation dose (LD_{30}) that causes 30% reduction in growth using *in vitro* shoot tips was found to be 7.65Gy, which indicates different radiosensitivity should be established based on the planting materials used for mutation induction. In our observation, the effective mutation dose may be found below LD_{50}.

B. *Effects of gamma radiation in germination and growth of A.maquilingensis seeds*

The first signs of germination were observed 13 days after treatment for seeds irradiated at 10 Gy, 20Gy and 30Gy at a rate of 16.3%, 9.2% and 8.9%, respectively. On the other hand, unirradiated (control) seeds started sprouting on 16th day with 7.4% germination while seeds irradiated at 40Gy (11%) and 50Gy (10%) initiated germination on day 21(data not shown). As presented in Figure 2, the highest recorded germination rate of seeds three weeks after incubation was at 10 Gy (36%) while a significant decline of survival was observed in doses above 30 Gy. Nevertheless, germination rate of unirradiated seeds remained lower than the irradiated samples, which indicates that exposure of seeds to radiation can possibly break seed dormancy or enhance the seed development. In addition, no visible aberrations were observed yet for the irradiated *in vitro* seedlings, however, the number of roots produced by the samples irradiated at 10Gy was found to be significantly higher than control and 20Gy (Fig.4). On the other hand, germinated seeds irradiated at 40 and 50 Gy did not produce any roots nor shoots even after 2 months in culture (Fig.5). This indicates that low dose (10 Gy) can enhance the growth and development of *Alocasia maquilingensis* seeds. Lettuce seeds exposed to low radiation doses (2-30Gy) enhanced the growth parameters (final germination percentage, germination index, root and hypocotyl length) as compared to untreated plants [13]. In the same way, stimulated germination of tomato seeds was achieved after irradiation at 10Gy and eventually increased the fruit production up to 86% [14]. This phenomenon maybe attributed to the high production of reactive oxygen species (ROS) such as hydrogen peroxide (H_2O_2) during irradiation which in turn can result to the reduction of germination inhibitors [15].

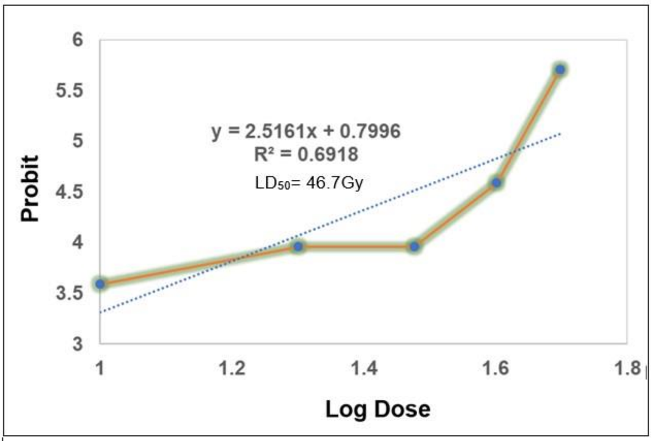

Figure 2. Plotted LD_{50} of *in vitro* cultures of *Alocasia maquilingensis* two months after treatment based on probit regression analysis

Dormancy is an important survival adaptation innate to most plants as a response to unfavourable environmental conditions. An important function of dormancy is delayed germination wherein seeds and seedlings are protected from suffering from harsh environments and herbivores; being activated only when competition from other plants for light and water might be less intense [16]. Most *Alocasia* species are seasonally dormant [1] with seeds tend to have low viablity upon exposure to dry condition and with a problematic storage properties[17]. In our

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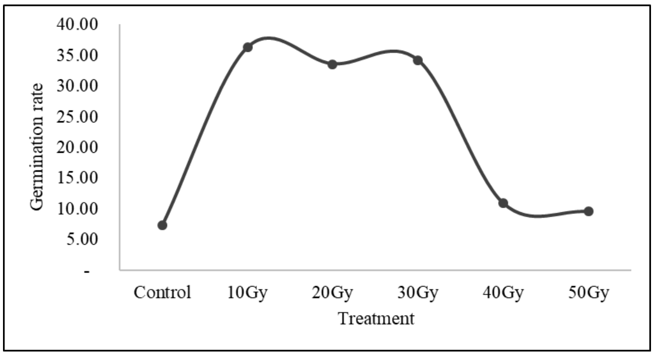
experiment, we performed aseptic culture of immature seeds (embryo rescue), which are considered mophologically dormant, thus, a low to non-existent germination was expected. However, irradiation at low dose (10Gy) did not only increased the germination rate of the immature seeds but also resulted to the increased root production which is a very important trait for survival. Enhancing the germination and viability of seeds is very important in germplasm establishment for reintroducing threatened plant species for ecology rebuilding [18]. Several reforestation efforts used low-dose irradiation for enhancing growth performance of forest species [19]. Therefore, the present study can be useful for conservation program for *Alocasia* and other aroids.

Figure 3. Germination rate of irradiated and non-irradiated seeds of



A.maquilingensis after 3 weeks incubation *in vitro*

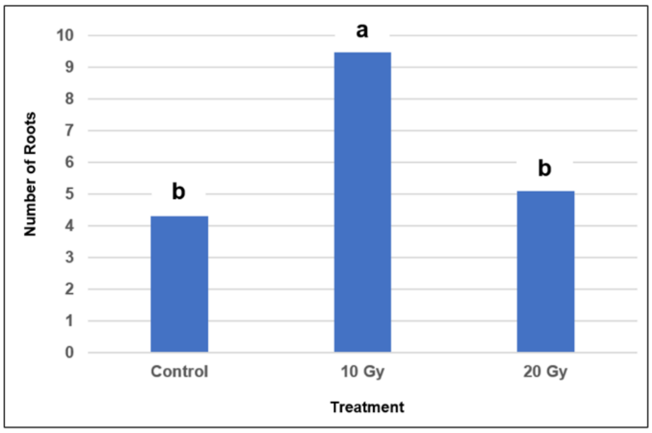


Figure 4. Average number of roots of seed derived *in vitro* plantlets 2 months after treatment.

Note:bars with the same letter are not significantly different at each other at p=0.05 (Tukey's LSD)

Figure 5. Germinated *in vitro* plantlets from immature seeds irradiated at 0, 10, 20, 30, 40, and 50 Gy after 2 months in culture

IV. CONCLUSION

To our knowledge, this is the first report describing the effects of gamma radiation in *Alocasia* species seeds. The median lethal dose (LD_{50}) was determined to be at 46.7Gy. No visible aberration was observed yet among irradiated samples but lower doses (i.e 10 Gy) of gamma rays improved growth rate performance of *A. maquilingensis* while higher doses (40 and 50Gy) delayed seedling growth. This indicates that using lower doses of gamma rays can break dormancy and enhance germination performance of *Alocasia* immature seeds. Our results can be useful for conservation activities for *Alocasia* and other aroids. In the meantime, *in vitro* irradiated seed-derived plantlets will be acclimatized and grown to full maturity and observe further mutations for developing new varieties of *Alocasia* with economic and ecological values.

ACKNOWLEDGMENT

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SyMo: An Australian First of A Kind Waste Treatment Plant Utilizing Synroc Technolgy

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I. INTRODUCTION

Synroc is an Australian technology which is being developed within the Australian Nuclear Science and Technology Organisation (ANSTO) for the treatment of radioactive waste. The SyMo Facility will implement Synroc technology which utilizes extreme temperatures and pressures within a Hot Isostatic Press (HIP) to immobilize radioactive elements within a high density crystalline lattice. The chemical constituents within this lattice have been designed to produce a final wasteform which mimics natural minerals that have demonstrated the capability to immobilize radioactive elements over geologically significant timeframes, providing a low risk solution for long term disposal due to the stability of the resulting non-leaching ceramic matrix [1]. This paper will present an overview of the SyMo Facility under construction at ANSTO which implements the Synroc technology, and provide a high level summary of the key Synroc processing steps.

II. SYMO FACILITY

The SyMo Facility commenced construction in 2018, and will be a first of a kind dedicated processing plant which utilizes Synroc technology to treat Intermediate Level Liquid Waste (ILLW) generated from the production of medical isotopes at the new ANSTO Nuclear Medicine (ANM) Molybdenum-99 production plant. The SyMo Facility will also treat ILLW currently held in storage tanks at ANSTO's previous Molybdenum-99 production plant, B54.

Figure 1. Conceptrual desgin of the SyMo Facility.

Figure 2. Construction of the SyMo Facility commenced in July 2018. The above image depicts the main Hot Cell Complex and structural steel portal frame which are currenlty nearing completion. Photoe dated December 2019.

III. INCOMING ILLW STREAM

The SyMo Facility houses a Synroc treatment process that has been tailored to treat an incoming stream of ILLW which is primarily ~6 M NaOH, with traces of $NaAlO_2$ [2]. The Alkaline ILLW produced within the ANM Molybdenum-99 plant is held in storage tanks to decay short-lived radioisotopes prior to transfer to the SyMo Facility via an underground connection between the two facilities.

Existing ILLW from storage tanks at ANSTO's previous Molybdenum-99 production plant, B54, can be transferred into the SyMo Facility through a separate waste transfer connection which allows for transport of ILLW between the two facilities via flask.

IV. PROCESSING

The ILLW is transferred from the SyMo Facility holding tank into a processing Hot Cell and mixed with Synroc additives which have been designed to incorporate the radioactive waste elements into the final wasteforms chemical structure. The slurry produced is dried and thermally treated to remove moisture and form a fine granular powder. The powdered wasteform is then dispensed into a canister and sealed for Hot Isostatic Pressing [3].

V. Hot Isostatic Pressing

Hot Isostatic Pressing is the simultaneous application of high temperature and high pressure to materials for a controlled duration of time in order to alter their mechanical properties [4]. The HIP unit consists of a high temperature furnace enclosed within a pressure vessel. The sealed canister containing the powdered wasteform is placed inside the furnace and heated with argon, which applies isostatic pressure uniformly in all directions [5]. The pressure, temperature and processing time are controlled in order to fuse the powder into a high density solid wasteform.

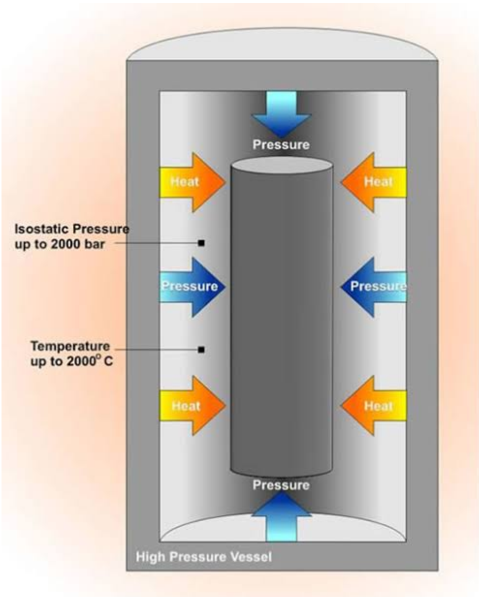

Figure 3. Schematic of the Hot Isostatic Pressing Process [6].

VI. Wasteform

After treatment with the Synroc process, the result is a chemically durable wastefrom which incorporates the radioisotopes from the ILLW within its ceramic matrix [1].

The canister includes a bellows design which allows for control of the canisters collapse profile during the HIP process, to ensure the final product is a compact cylinder to optimize for storage. The Synroc waste form has a high waste loading compared to conventional technologies [2], which allows for increased space efficiency for long term storage.

The Synroc wasteform has been designed to meet international requirements for long term disposal [7]. The combined durability over geologically significant time frames and space efficiency make Synroc a safe, secure and sustainable wasteform for long term disposal.

Figure 4. Representation of a pre HIP canister (back left). During the HIP process, the canister is designed to collapse into a cylindrical shaped vessel (right) to aid storage. The centre image represents a cross-section of the dense solid waste form that is encapsulated within the canister [3].

VII. Key Challenges Going Forward

The SyMo Facility is forecast to complete construction and fit out in early 2021 and will be followed by a commissioning period to demonstrate the functionality and safety of the processing facility. As a first of a kind plant, commissioning and process integration presents a number of challenges. To mitigate and overcome these challenges before they eventuate, ANSTO have constructed in parallel a demonstration plant which implements key processing steps on a one-for-one scale with the end plant, and are currently undertaking an extensive series of trials to demonstrate robustness, investigate optimization options, and train future staff in operation and maintenance of key equipment.

VIII. Conclusion

Upon completion of commissioning, the SyMo Facility will be the first dedicated waste treatment plant to utilize the Australian developed Synroc Technology, and represents a key milestone for innovation within the Australian and global nuclear industries. The completion of this project will demonstrate the feasibility of the technology at a commercial scale, a key milestone for ANSTO, as Synroc technology has commercial viability for tailoring to a variety of waste streams.

The SyMo Facility allows ANSTO to provide a sustainable and conscionable supply of nuclear medicine to domestic and international markets, by ensuring safe and responsible conversion of Intermediate Level Liquid Waste generated from the production of Molybdenum-99 into a durable, space efficient wasteform which has been designed and optimized for long term disposal.

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IYNC2020

CLOSING REMARKS

Dear IYNC2020 participants,

It has been a great pleasure to see how the IYNC2020 conference edition has again succeeded in bringing together such a significant number of young professionals from all around the world.

Overall, more than 250 senior and young nuclear professionals from over 42 countries have shared their views on the key topics linked to nuclear science and technology thanks to outstanding and inspiring speeches in dedicated Plenary Sessions, high-level Technical Presentations as well as Panel and Workshop discussions.

This edition has been particularly interesting due to its complementary view on non-power applications, in consonance with the nuclear sector in Australia. Indeed, we believe that the conference has fulfilled its goal of raising awareness on how the diverse and peaceful uses of nuclear science and technology can improve our lives.

All things considered, IYNC2020 has achieved its mission, and this has been possible thanks to you, the participants, for your eagerness to share your views and connect with each other during the conference. Please receive our deepest gratitude.

Finally, we would like to take this opportunity to sincerely thank all the people who have been involved in the organization of the Conference for their commitment and generous dedication to make of this conference such a success. Without them, none of this would have been possible.

These conference proceedings represent the end of this exciting journey that the organization of IYNC2020 has provided, but also the beginning of new adventures to come within the frame of future IYNC conferences.

We hope to meet you there again all at one!

What is past, is prologue

Kevin Fernández-Cosials	Ignacio Gómez-García-Toraño
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