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Considerations for the Back End of the Fuel Cycle of Small Modular Reactors

Proceedings of a Technical Meeting

IAEA
International Atomic Energy Agency

CONSIDERATIONS FOR THE BACK END OF THE FUEL CYCLE OF SMALL MODULAR REACTORS

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IAEA-TECDOC-2040

CONSIDERATIONS FOR THE BACK END OF THE FUEL CYCLE OF SMALL MODULAR REACTORS

PROCEEDINGS OF A TECHNICAL MEETING

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FOREWORD

There is an increasing interest in small modular reactors and their applications, in part owing to their consideration as a low carbon energy source in the climate change mitigation plans of many Member States. According to the IAEA material on Advances in Small Modular Reactor Technology Developments, complementary to ARIS database, there are more than 80 small modular reactor concepts currently under development, spanning a significant range of reactor technologies.

Small modular reactor concepts include evolutionary variants of light water reactors, that benefit from many decades of operating experience with the current fleet of nuclear power plants; high temperature gas reactors; liquid metal fast reactors; and molten salt reactors. Small modular reactor designs use a variety of fuel forms (e.g. oxide/ceramic, metal, tristructural isotropic particle and prismatic fuels and liquid fuel salts) having different fuel compositions and forms, for example uranium oxide (low enriched uranium, high assay low enriched uranium); mixed uranium and plutonium (oxide, metal or salt); and kernel particles.

While much focus has been given to aspects of the deployment of small modular reactors, such as reactor concepts, engineering, economics, infrastructure, the fuel cycle and in particular the management of the spent nuclear fuel appears to have had limited consideration so far. As small modular reactor concepts are becoming more refined, it is an appropriate time to start identifying the challenges, opportunities, gaps and issues for managing spent fuel from small modular reactors.

In this regard, a Technical Meeting on Considerations for the Back End of the Fuel Cycle of Small Modular Reactors was organized in September 2022 to facilitate the exchange of information and discussions to enable experts to collaboratively identify the opportunities and challenges faced at all stages of the back end of the fuel cycle (e.g. storage, transportation, reprocessing and recycling, and disposal), the gaps in current infrastructure and the knowledge required to ensure an integrated approach to the overall spent fuel management strategy, as well as the potential ways to move forward in addressing them in the near, medium and long terms.

The Technical Meeting was attended by 107 participants from 32 Member States, and 3 international organizations, delivering 40 oral presentations scheduled in three technical sessions on IAEA perspectives, international perspectives, and Member States' perspectives. Three topical breakout sessions on small modular reactors based on light water reactors, on high temperature gas cooled reactors and on advanced reactors (liquid metal cooled and molten salt) were held. This publication presents the proceedings, session summaries and conclusions of the Technical Meeting as well as the 26 extended abstracts and full papers presented at the meeting.

The IAEA expresses its appreciation to all contributors to this publication. The IAEA officer responsible for this publication was A. González-Espartero of the Division of Nuclear Fuel Cycle and Waste Technology.

CONTENTS

1. INTRODUCTION

1.1 BACKGROUND

There is an increasing interest in Small Modular Reactors (SMRs) and their applications, in part due to their consideration as a low carbon energy source in the climate change mitigation plans of many Member States. SMRs are newer generation reactors designed to generate electric power typically up to 300 MW(e) and for non-electrical industrial applications (e.g., water desalination and heat generation for industrial processes). According to the IAEA Booklet on Advances in SMR Technology Developments [1], complementary to ARIS and published in 2022, there are more than 80 SMR concepts currently under development, spanning a significant range of reactor technologies. The SMR concepts can be deployed in a variety of configurations, ranging from single-unit installations to multimodule plants with a different degree of modularization across designs to suit the requirements of the operator.

SMR concepts vary from evolutionary variants of Light Water Reactors (LWR-SMRs, either land or marine based), that benefit from many decades of operating experience of the current fleet of LWRs; High Temperature Gas Cooled Reactors (HTGR-SMRs); Liquid Metal Fast Reactors (LMFR-SMRs) and molten salt reactors (MSR-SMRs). SMR designs use a variety of coolants (e.g., water, liquid metal, molten salts) and fuel forms (e.g., oxide/ceramic, metal, TRISO, liquid fuel salts.) having different Technology Readiness Levels (TRLs) and fuel compositions and forms (e.g., UOX (LEU, HALEU); Mixed U and Pu (oxide, metal, or salt); kernel particles).

While much focus has been given to aspects of SMR deployment such as reactor concepts, engineering, economics, infrastructure, safety, etc., the fuel cycle, and in particular the management of Spent Nuclear Fuel (SNF), appears to have had limited consideration. As the SMR concepts are becoming more refined, it is an appropriate time to start identifying the challenges, opportunities, gaps, and issues for managing spent fuel from SMRs during all stages of the back end of the fuel cycle such as storage, transportation, reprocessing & recycling, and disposal.

The management of spent fuel is very dependent on the characteristics of the nuclear fuel relating to its enrichment, matrix and composition, and its irradiation history (e.g., burnup). Spent fuels coming from SMRs will have different characteristics and irradiation histories that will require either adaptation of currently implemented technologies or new developments for all stages of the back end of the fuel cycle to accommodate higher thermal outputs and criticality risks, different radionuclide inventories, new matrices, and cladding behaviours, etc., implying the need for R&D, demonstration projects and licensing to ensure that the fundamental safety objective is met.

In its 17th Meeting in April 2019, the Technical Working Group on Nuclear Fuel Cycle Options and Spent Fuel Management (TWG-NFCO) recommended that "*The next update of the Advances in Small Modular Reactor Technology Developments report should consider technologies for managing SNF from SMRs. This update should not only consider the reactor technology, but also the backend infrastructure that would be needed to support SMRs' deployment - including transportation, storage, recycling, and disposal technologies. Newcomer countries should be made aware that, as with all reactor types, the management of SNF from SMRs needs to be fully considered. Nuclear fuel cycle aspects, in particular the backend, should be integrated into all IAEA working groups that are looking at SMRs*".

Based on those recommendations, a Technical Meeting on Considerations for the Back End of the Fuel Cycle of SMRs was convened in September 2022 in Vienna. The purpose of the meeting was to facilitate the exchange of information and discussions regarding the management of spent fuels coming from all envisaged SMR technologies to enable experts to collaboratively identify the opportunities and challenges faced at all stages of the back end of the fuel cycle (e.g., storage, transportation, reprocessing & recycling, and disposal), the gaps in current infrastructures and the knowledge required to ensure an integrated approach to the overall spent fuel management strategy, as well as the potential ways to move forward in addressing them in the near, medium, and long terms.

The technical meeting was scheduled in three technical sessions on (i) IAEA Perspectives, (ii) International Organizations Perspectives, and (iii) Member States Perspectives. Three topical break out sessions on LWR-SMRs, HTGR-SMRs and Advanced Reactors including Molten Salt Reactors (AR(MSR)-SMRs) were held.

The meeting was attended by 107 participants from 32 Member States, and 3 International Organizations, delivering 40 oral presentations. The attending countries and organizations were Argentina, Armenia, Belarus, Plurinational State of Bolivia, Bulgaria, Canada, China, Czech Republic, Egypt, Ethiopia, Finland, France, Hungary, India, Japan, Jordan, Lithuania, Malaysia, Netherlands, Pakistan, Philippines, Poland, Romania, Russian Federation, Singapore, Slovenia, South Africa, Sudan, Sweden, Thailand, United Kingdom of Great Britain and Northern Ireland, United States of America, Uzbekistan, the European Commission (EC), the OECD/Nuclear Energy Agency (NEA) and the European Repository Development Organization (ERDO).

1.2 OBJECTIVE

The objective of this publication is to provide the Member States a reference, with the proceedings, discussions, session summaries and conclusions of the Technical Meeting, including the extended abstracts and full papers presented at the meeting.

1.3 SCOPE

This TECDOC presents the Proceedings of the Technical Meeting on Considerations for the Back End of the Fuel Cycle of SMRs held in September 2022. This publication compiles summaries of the technical sessions, group discussions, and conclusions as well as 26 extended abstracts/full papers submitted to and presented at the meeting.

1.4 STRUCTURE

Section 1 provides the introduction to the publication. Section 2 summarizes the three technical/meeting sessions. Section 3 provides a summary of the discussions during the break out sessions and Section 4 condenses a recap of the general discussion. Section 5 highlights main summary and conclusions from the meeting. Extended abstracts and full papers submitted to the meeting are also included in this publication in the order they were presented according to the Meeting Programme.

2. SUMMARY OF MEETING SESSIONS

This chapter includes a summary of presentations delivered in the three Technical Sessions as well as the main topics of discussions. According to the Meeting Agenda, reflected in the Appendix, Session I was dedicated to ongoing activities that the IAEA is conducting in the SMRs field, with nine contributions from the Departments of Nuclear Energy, Nuclear Safety and Security, and Safeguards. Session II included information on International Organizations activities on SMRs with contributions from EC, OECD/NEA and ERDO. Session III comprised Member States' activities on SMRs with 27 presentations.

2.1. SESSION I: IAEA PERSPECTIVES

Prepared by Mr Surik Bznuni (Armenia, Nuclear and Radiation Safety Center)

Paper ID#3 IAEA (NPTDS, Nuclear Power Technology Development Section), 2022 IAEA SMR ARIS Booklet on Advances in SMR Technology Developments – Mr Hadid Subki

Mr Subki presented characteristics and attributes of SMRs and the status of advances on SMR technologies. Typically, SMRs are advanced reactors that produce up to 300 MW(e), built in factories and transported as modules to sites for installation as demand arises. The key attributes of SMRs are six: economic, modularization, flexible application, smaller footprint, can be replacement of ageing fossil-fired plants and can be potentially part of hybrid energy systems. SMRs can be classified in various ways according to technology and capacity. The IAEA has summarized the current status of these SMR developments in the IAEA ARIS (Advances in Small Modular Reactor Technology Developments) SMR Booklet, 2022.

A subset of SMRs are microreactors (1–20 MW(e) capacity) that have common features with the rest of SMRs. The following specific characteristics and applications of microreactors were considered in the booklet:

- Inherent and passive safety features;
- Substantially lower upfront capital costs;
- Much smaller footprints, reduced-sized or even eliminated EPZ (Emergency Planning Zones);
- Rapid deployability due to modularity (even an entire reactor);
- Spent fuel smaller in size;
- Scalability, resiliency, self-regulating;
- Potential to operate in island-mode & to black-start;
- High transportability from mobility;
- Long refuelling interval.

The IAEA ARIS SMR booklet 2022 [1] includes not only information on the technical aspects of SMR designs, but also some considerations for fuel cycle approaches, waste management and disposal plan by SMR type.

Paper ID#4 IAEA (NPTDS, Nuclear Power Technology Development Section), The IAEA Platform on SMRs and their Applications: Progress and Achievements after one year – Mr Stefano Monti

Mr Monti introduced the IAEA's Platform on SMRs and their Applications and elaborated on the progress of its development after one year. This Agency-wide platform has been developed from Member States requests to coordinate and optimize the Agency effective and efficient support to Member States, international organizations and stakeholders dealing with SMRs and willing to cooperate with the IAEA.

The first achievement is the actual and effective coordination of all Agency activities on SMRs and their applications throughout the relevant Departments and Offices. The second was to establish a medium-term strategy thorough analysis of the ongoing and planned IAEA projects, initiatives, and activities on SMRs and their applications. The third was to prepare the high level 'SMR Booklet: A New Nuclear Energy Paradigm', currently available at the IAEA online preprint repository. The target audience for this material is policy makers and government officials interested in SMRs. The fourth achievement was to create the SCORPION SMR portal that gathers external information on SMRs. The fifth achievement was the establishment of the new TC-Interregional INT2023 Project (2022–2025). The main objective of this project is to improve technical knowledge, capacity building and safety review capability in developing countries addressing the fundamental aspects of SMRs and their electric and non-electric applications. The sixth achievement was the creation of Agency-wide task forces to work in specific activities related to SMRs.

An SMR platform annual report was delivered for the period 2021–2022 summarizing the relevant work carried out and managed by all IAEA Sections working on SMRs and their applications.

Paper ID#5 IAEA (NFCMS, Nuclear Fuel Cycle and Materials Section), IAEA Ongoing Activities on Spent Fuel Management – Ms Amparo González Espartero

Ms González Espartero presented ongoing IAEA activities to address spent fuel management challenges, that can be summarised as the increasing of storage periods that requires ageing management programmes to be put in place to confirm spent fuel integrity and to ensure that the systems, structures and components maintain their safety functions; transportability of storage packages after long storage periods; implementation of multi-recycling of U/Pu in LWRs at industrial scale; demonstration and scale-up of multi-recycling through advanced fuel cycles for innovative reactors (Gen-IV); and to accommodate the management of new spent nuclear fuels (e-ATFs and SNF from SMR technologies).

The IAEA is conducting Coordinated Research Projects (CRPs) to address those issues, some of them are already closed and documented ('Demonstrating Performance of Spent Fuel and Related Storage System Components During Very Long Term Storage' TECDOC-1878; 'Spent Fuel Performance Assessment and Research' (SPAR IV) TECDOC-1975, and 'Ageing Management Programmes for Dry Storage Systems', TECDOC under preparation) and others are active and still open for proposals ('Spent Fuel Characterization' T13018, 'Spent Fuel Research and Assessment (SFERA)' T13020, and 'Performance Assessment of Storage Systems for Extended Durations (PASSED)' T13019).

The IAEA 'Guidebook on Spent Fuel Storage Options and Systems, 3rd Edition' IAEA-TRS-240, has been recently made publicly available through the IAEA Preprint Repository in 2022 and gathers information on current spent fuel storage systems and options, describing the main features of storage systems, showing the distribution of the current SNF inventory in the different storage systems by regions and by countries, a harmonised scheme of dry storage systems and examples of the corresponding commercially available systems.

Detailed information on these and other IAEA activities on SNF recycling, IAEA International Conferences, e-Learning Course on spent fuel management and IAEA webinars on spent fuel management can be found at the IAEA Spent Fuel Management Network (SFM Net) public page.

Paper ID#6 UK (NNL), Information on the IAEA TM on Back End Opportunities and Challenges for Spent e-ATFs Management – Mr David Hambley

Mr Hambley presented a summary of an IAEA Technical Meeting held virtually back in June 2022 which was focused on back end opportunities and challenges for managing spent Evolutionary Accident Tolerant Fuels (e-ATF). Some of those fuels are likely to be deployed in the current fleet of reactors and in some of the different SMR technologies in the near term. E-ATFs can be licensed under existing regulatory guidelines and typically involve fuel for LWRs (large or small); coated zircaloy fuel (e.g., FeCrAl) and potentially doped UO_2.

The Technical Meeting pointed out that there is a lot of work underway to understand the impact of e-ATFs on back end activities. There was a common recognition of the need for irradiated fuel characterization data and testing to validate the models that support the back end fuel cycle stages. The importance of multilateral cooperation was remarked upon by several participants.

Currently, there are lead assemblies under test irradiation in multiple reactors and various industrial implementations were proposed such as batch loading which represents a significant shift from lead test assemblies and qualified commercialized manufacturing process. For storage and transportation, data is needed to support the use of proven methods for criticality, shielding, etc., fatigue, mechanical behaviour and long term degradation of coated claddings and dryness determination.

For reprocessing, shearing behaviour of chromium coated cladded fuel needs investigation to determine whether additional fines are produced. Routing of the fines and the impact on plant with respect to corrosion and PUREX performance need to be determined. The presence of chromium in raffinate may impact vitrified glass quality with impact on disposal routes. The meeting participants came up with some recommendations for future IAEA activities, such as the organization of a workshop to identify needs and collaborative R&D opportunities to underpin spent e-ATF management.

Paper ID#7 IAEA (NSNI, Division of Nuclear Installation Safety), Consideration of Non-Water Cooled Reactors and SMRs in the IAEA Safety Standards – Ms Paula Calle

Ms Calle presented results of the analysis of the applicability of the IAEA safety standards to non-water cooled reactors and SMRs. The approach to conduct the review was to identify areas of novelty compared to LWRs, gaps as well as areas where current safety standards might not be applicable.

The following high-level conclusions were highlighted:

- Some safety approaches focus on conventional LWRs and do not cover SMR specificities;
- There is lack of experience on practical application;
- First of a kind (FOAK) issues are not properly considered;
- New modes of failures, equipment failures or phenomena are not covered.

In Construction and Manufacturing, advanced manufacturing and factory-based aspects need further consideration. In Commissioning and Operation, Accident Management areas of alternative operating models and implications from novel and FOAK need further consideration. In Leadership and Management for Safety areas, oversight of manufacturing if operator is not known and novel features implications on management system need further consideration. In Legal and Regulations area, international cooperation, and impact of deployment model on regulatory oversight need further consideration.

In Fuel Cycle Facilities area, the Safety Standards cover current fuel fabrication processes, but additional Safety Guides would need to be developed to cover the fabrication of advanced fuels and reprocessing of non-WCR fuel once there is sufficient knowledge and experience on these processes. Until such a point is reached, the guidance in SSG-43 is seen as sufficiently general to guide the safety of nuclear fuel cycle facilities associated with evolutionary and innovative designs.

IAEA is planning to develop new safety guide 'Safety Demonstration of Innovative Technology in Power Reactor Designs' to address FOAK issues.

Paper ID#8 IAEA (NSNS, Division of Nuclear Security), Security Considerations for Back-End of Nuclear Fuel Cycle for SMRs – Mr Tariq Majeed

Mr Majeed presented the IAEA perspective on security considerations for back end of nuclear fuel cycle for SMRs. General conclusion is that for LWR-SMRs back end security considerations are mainly the same as for conventional LWRs. Innovative SMRs with exotic design solutions, like molten salt reactors could pose specific challenges on providing security. Development of new IAEA technical documents related to security of SMRs are in progress.

Paper ID#9 IAEA (SGCP, Division of Concepts and Planning), Safeguards Considerations for SMR Fuel Cycles – Mr Kerrin Swan

Mr Swan presented safeguards considerations for SMR fuel cycles. It was highlighted that all SMRs and related nuclear fuel cycle facilities, built in States with comprehensive safeguards agreements – even prototypes – will need to be safeguarded, regardless of the size, technology, or State of origin.

The following main challenges for SMRs safeguards were identified:

- New fuels and fuel cycles: Th/U-233, RepU, MOX, TRU fuels, pebble bed, prismatic core, pyroprocessing, and other new processes;
- New reactor designs: molten salt, fast neutron, micro-sized, etc;
- Longer operation cycles: continuity of knowledge of nuclear material needs to be maintained (for example, between refuelling), high excess reactivity of core which could be used for target accommodation;

- New supply arrangements as factory sealed cores, transportable power plants, transnational arrangements (need for design verification at State of origin and receiver State, IAEA sealing);
- New spent fuel management: storage configurations, waste forms;
- Diverse operational roles: district heating, desalination, hydrogen and electricity;
- Remote, distributed locations: access issues, lack of "unannounced" visit deterrence, cost-benefit issues.

Therefore, important safeguards features will be needed, such as unattended monitoring systems and remote data transmission; digital connectivity to coverage in remote areas (reliable, high bandwidth, secure); safeguards seals on factory-sealed and transportable cores; design verification, particularly under transnational supply arrangements; and, new safeguards approaches, including (potentially) customized IAEA or joint-use instrumentation. Other safeguards considerations are managing the efficient and effective implementation of safeguards for fleets of small, remotely distributed SMRs.

One of the important design features that IAEA pursues is safeguards by design that integrates safeguards considerations into the design process of new or modified facility, at any stage of the nuclear fuel cycle, from initial planning through design, construction, operation, waste management and decommissioning. This is a voluntary process that neither replaces a state's obligations for early provision of design information under its safeguards' agreement, nor introduces new safeguards requirements.

Paper ID#10 IAEA (PESS, Planning and Economic Studies Section), CPR on the Economic Appraisal of SMRs – Mr Saied Dardour

Mr Dardour presented the preliminary results of the IAEA CRP on the economic appraisal of SMR projects. Within this project a generic methodology will be developed for assessing the relevance of the SMR option in a given context and for demonstrating the business case for SMRs. It is also planned to develop country cases and other case studies, focusing on SMR applications, and illustrating the implementation of the suggested methodology.

Paper ID#11 IAEA (NSRW, Division of Radiation, Transport and Waste Safety), Transport Safety and Back End of SMR Fuel Cycle – Ms Shazia Fayyaz

Ms Fayyaz presented the IAEA perspective on transport safety for the back end of SMRs fuel cycle. It was concluded that SSR-6 which covers transport of radioactive material considering classification of material and packages remains applicable if cargo approach is considered for the back end of SMRs fuel cycle. However, the concept of transportable reactors is out of the scope of current IAEA transport safety framework.

2.2. SESSION II: INTERNATIONAL PERSPECTIVES

Prepared by Mr Surik Bznuni (Armenia, Nuclear and Radiation Safety Center)

Paper ID#12 EC/JRC, JRC Contribution to SMRs and the Back-End Fuel Cycle – Ms Concetta Fazio

Ms Fazio presented EC/JRC contribution to SMRs and related back end of the fuel cycle. The JRC mission and vision as well as the key objectives of the EURATOM framework programme

2021–2025 were presented, followed by the introduction of the SMRs in a European context and listed the main expectations and challenges for this new type of reactors.

The issues to be considered for the SMRs' waste management in terms of pre-disposal and disposal phases, are related to the different types of fuels, enrichment, burn-up, cladding materials and applied fuel cycle. To anticipate SMRs' waste management studies, the JRC performs investigations on spent nuclear fuel/waste analogues. The studies are aimed at providing data that would allow the confirmation of the compatibility with existing/planned waste management schemes. The investigations are oriented towards the pre-disposal and final disposal stages. The final disposal studies, which are relevant for the safety case, include corrosion experiments to correlate the irradiation history with the instant release fraction; the assessment of the long-term stability of the SNF in reducing/oxidizing conditions and addresses also the SNF heterogeneity. Studies related to the pre-disposal stage are aimed also at measuring the response of SNF to different mechanical loading conditions.

In conclusion, there is a growing interest in Europe towards SMRs and there are several SMR designs under consideration. Specific aspects that can possibly affect the back end of SMR fuel cycles needs to be identified along with R&D focusing on these aspects through dedicated studies.

Paper ID#13 OECD/NEA, Update on OECD/NEA Activities Regarding the Back end of SMRs – Ms Rebecca Tadesse

Ms Tadesse presented an update of OECD/NEA On-going Activities on the Back End of the Fuel Cycles for SMRs. The results of the ad-hoc OECD/NEA Expert Group on Extended Storage and Transportation were discussed, highlighting that:

— There exists a wide range of specialized tools and analyses for certain areas of the fuel cycle (looking at optimized operation, optimized loading of final disposal canisters with respect to maximum heat per canisters, etc.);
— Clarity with respect to the end point of SNF storage is needed. Storage (even extended) is an interim stage of RW/SNF management, and it is necessary to identify the appropriate timescales and the purpose of RW/SNF storage;
— From the point of view of organizational framework, it is necessary to ensure that the extended storage does not lead to the appearance of a new legacy for next generation.

Paper ID#14 ERDO, The Potential Impacts of SMRs on Multinational Cooperation at the Back end of the Fuel Cycle – Mr Charles McCombie

Mr McCombie presented ERDO association's activities on analysis of the back-end options for SMRs. Acceptance of nuclear is/has been strongly affected by disposal issues, therefore, ERDO looks for multinational cooperation in addressing challenges of the back end of the fuel cycle. The ongoing US Department of Energy sponsored project on the potential impacts of SMRs on multinational cooperation at the back end of the fuel cycle was discussed.

Key messages of his presentation were:

— If the 'waste disposal problem' is removed by a take-away offer, then non-nuclear countries might reconsider the nuclear option. New nuclear countries are more likely to order an SMR if the supplier takes back the entire module or the SNF;

- SMRs may enhance the ‘image’ and the acceptability of nuclear power so that large NPPs also become more acceptable;
- Multiple SMR customers of the same design may cooperate on developing SNF conditioning and packaging approaches;
- Suppliers of SMRs – especially those with novel fuel cycles – may be interested in building multinational ‘user groups’;
- Major established disposal programmes may see opportunities in accepting relatively modest amounts of SNF from new SMR countries;
- Suppliers of SMRs may exert pressure on their home countries to accept return of spent core modules or of SNF elements.

2.3. SESSION III: MEMBER STATES PERSPECTIVES

Prepared by
Mr Andrea Salvatores (France, Commissariat à l'Énergie Atomique et aux Énergies Alternatives)
Mr David Hambley (UK, National Nuclear Laboratory)
Ms Fatimah Al Momani (Jordan, Energy and Mineral Regulatory Commission)

Paper ID#16 Armenia (NRSC), Safety Implications on the Back End of the Fuel Cycle for SMRs – Mr Surik Bznuni and Mr Vahram Petrosyan

Armenia operates a single nuclear power station and a dry storage facility based on the NUHOMS concept. According to the country’s current strategic plan for energy sector development (2015–2036) a nuclear power unit needs to necessarily be part of the electricity generation installation structure. In this framework, activities are foreseen for second lifetime extension of the existing nuclear power unit to permit operation for an additional 10 years (2026–2036). Before 2036 a new nuclear power unit, as replacement capacity, needs to be commissioned. Additionally, in 2023 it is planned to extend the dry storage facility.

Concerning the reactor choice, currently three possible technological options could be realistic for Armenia: a mid-size PWR unit with about 600 MW(e) capacity or 2 small size PWR units with about 300 MW(e) capacity each or several SMRs. Regarding this last type of reactor, SMR designs consider the use of fuel with enrichment higher than 5% threshold. This enrichment could have significant implications on the back end of fuel cycle. Indeed, LWR fuel analysis were validated mainly based on experiments with enrichments of less than 5%. Therefore, the validation of models and codes might be extended. New experiments involving both fresh and spent fuel assemblies with enrichments higher than 5% have to be carried out to support this validation. These considerations are not limited solely to neutronic core calculations but also to transport and storage.

These new fuels will have a strong impact on the design of the transportation cask because of criticality considerations. The limitation of the number of fuel assemblies and burnup credit could be used to tackle these issues. The residual decay heat arising from increased irradiation will also affect the storage. The thermomechanical properties need to be addressed and re-evaluated by designers and vendors in terms of corrosion, hydrogen uptake, and volatile fission products in the spent fuel.

Paper ID#17 Uzbekistan (The State Committee of Industrial Safety of the Republic of Uzbekistan), Fuel Cycle Consideration for WWR-SM research reactor – Mr Boltabaev Azizbek

Uzbekistan has six nuclear facilities used for R&D. The research reactor fuel evolved by gradually lowering the uranium-235 content which results in an increase of the number of assemblies in the core.

Initially the number of assemblies was 18 and increased to 20 and finally to 24 assemblies. Managing the storage of spent assemblies is therefore a key issue. Uzbekistan has changed its storage capacity, both for fresh and spent fuel. Storage capacity has been recovered during recent years by means of a number of spent fuel exports. The storage of all fuels (fresh and spent) is accompanied by a physical protection system consistent with the management of these materials.

Paper ID#18 Egypt (EAEA), Direct Recycling of SMR Spent Fuel for Uranium Utilization Improvement and Reduction of High-Level Nuclear Waste – Mr Nader M. A. Mohamed

Due to of its small core volume, an SMR core has a higher neutron leakage than that of large-scale reactors. It follows that SNF discharged from SMRs will present a proportionally higher residual content of fissile materials which would: (1) degrade the uranium utilization and (2) increase the quantity of high-level waste. Therefore, direct recycling of the SMR spent fuel in CANDU reactors, through for instance the DUPIC cycle (originally proposed in Republic of Korea), would improve the uranium utilization and reduce the high-level waste.

As a case study, direct recycling of the spent fuel of NuScale SMR was investigated. In this case study, NuScale SMR is loaded with fuel that has an average enrichment of 4.175% U-235 and discharges the spent fuel with average burnup of 40.4 GWd/t. Calculations using MCNPX computational code showed that recycling the NuScale SMR spent fuel in CANDU-6 reactor would give additional burnup of 20 GWd/t, increasing the burnup in about 50%. Also, the high-level waste will be reduced by the same percentage i.e., 50%. Calculations of the evolution of the reactor cores provided an indication that SMRs could be integrated in an efficient and optimized manner in a large reactor fleet, not only in terms of the production of electricity but also of their back end management.

Paper ID#19 Slovenia (Slovenian Nuclear Safety Administration), Design Extension Conditions for Spent Fuel Storage at the PWR NPP KRŠKO – Mr Tomi Živko

Slovenia is the smallest country with a nuclear power programme. The storage of spent fuel is a major subject for the sustainability of nuclear activities and two main factors had impacted its management during the last decades. On one hand, the Fukushima accident, which gave raise to new considerations regarding the safety of storage, and on the other hand, the extension of the operating life of the NPP, led to changes in the SNF storage options with the creation of dry storage technologies and the improvement of SNF pool safety, through the installation of a mobile cooling system (firefighter or river) and a dry storage design resistant to earthquakes and severe atmospheric events. The end of the construction of spent fuel dry storage is planned for 2022 and loading is planned to commence in 2023. An important point of these evolutions is the fact that the specifications of the Slovenian regulations in principle make it possible to receive the storage of SMR fuels — although some points would need to be validated. Once again, it is noted that the progression of nuclear power has an impact not only on the reactors but also on the various facilities of the fuel cycle.

Paper ID#20 Poland (INCT), The Process of Decarbonization of the Domestic Power Industry in Poland using SMR Reactors – Ms Agnieszka Miskiewicz

A comprehensive presentation of the Polish energy panorama highlighted the importance of the fossil component in its energy mix and its need to evolve towards de-carbonized energies. Poland does not have a nuclear power reactor but 6 LWR units are planned to commence operation by 2033. In addition, low power HTGR reactors (200–350 MW) are considered for heat generation. This decision on HTGR would be consolidated by the construction of a small pilot unit of 10 MW(th). This panorama is expected to be completed by the evaluation of several SMR concepts. Work is underway to develop criteria for evaluating SMR suppliers individually as well as by a set of criteria bearing on a more global approach to address needs for a fleet of reactors.

Paper ID#21 Jordan (JAEC), Considerations and Perceptions for Nuclear Fuel Cycle Back End Related to the SMRs under Consideration in Jordan – Ms Fatmah Al Momani

Jordan, as in many other countries, is facing a growing demand of energy. Additionally, there is a real need for reliable and affordable domestic base load power. Available energy options are limited and have intrinsic limitations (durability, efficiency, environmental considerations, etc.). In this framework, nuclear energy is under consideration with two approaches at a different degree of maturity: large reactors (negotiations with vendors) or small power plants (technical and economic assessment). Concerning the SMR option, several factors are analyzed including integration in the current infrastructures, diversification of uses, safety, etc. In terms of the back end, after storage at reactor site, different technical options will be assessed, either considering the spent nuclear fuel as strategic resource that can be utilized through reprocessing (nationally or internationally) or declaring it as a radioactive waste to be disposed of directly in a national waste disposal facility. Until now no final decision on the spent nuclear fuel management option has been taken as it will be based on the selected reactor technology; three SMRs have been shortlisted based on matrix evaluation criteria. The management of the spent TRISO or oxide fuels is evaluated step by step including a potential reprocessing option outside Jordan.

Paper ID#22 France (CEA), Back End of the Fuel Cycle Considerations for Small Modular Reactors – Mr Andrea Salvatores

To reach carbon neutrality by 2050 to limit global warming, an energy transition has been initiated in France. Renewable and nuclear energies can play a significant role in a context of electrification of many uses. Nuclear energy development requires construction sites, financial commitments of several billion euro, societal approval and its delivery is complex. Therefore, small nuclear reactors, which require lower financial commitments, with shorter construction times and with greater simplicity to operate ought to be part of the French energy strategy. In the frame of the investment plan 'France 2030', €500M will be assigned to the development of the French SMR – Nuward and additionally, €500M will be dedicated to proposals for new reactor concepts in the field of fission and fusion. The main objective is to create a new ecosystem for the nuclear sector. The collaboration of the French nuclear safety authority with the Finnish and Czech nuclear safety authorities for a joint preliminary review of the Nuward reactor project is a good example of this new ecosystem. France has the main facilities needed for managing spent fuel. Evolutions/adaptations could be needed to accommodate new fuels coming from SMR concepts. A global vision of the nuclear fuel cycle is needed (starting materials and natural resources, life cycle, economy, storage, transportation, waste management).

Paper ID#23 France (Orano), Advanced Nuclear Reactors: What about the Back end? Focus on Treatment and Reprocessing/Recycling Aspects – Mr Renaud Liberge

The management of the back end of Advanced Reactors can be approached through their fuel types (Oxide/ceramic fuels with cladding, TRISO fuels, Metallic fuels, Liquid salt fuels) and fissile feature (LEU, LEU+, HALEU, HEU, mixed Uranium and plutonium, thorium). In the case of oxide fuels, (sintered pellet UO_2 or MOX fuel similar in design to the existing LWR oxide fuel), they can benefit from a consolidated operating, manufacturing, recycling and irradiation experience. The situation of the TRISO fuel is a little bit different due to the intrinsic low density of the fissile material in the fuel and the complexity to access it. So, a treatment of the spent TRISO fuel to remove/reduce graphite content would be needed for storage and for disposal. The pulsed currents could be a solution to separate the graphite component, maintaining the integrity of the TRISO particles. Metallic fuels are generally treated by pyroprocessing and aqueous polishing processes. A treatment for the residual sodium could be needed for storage and disposal steps. Finally, liquid fuels coming from Molten Salt Reactors could be treated by pyroprocessing or by integrating them in a treatment-recycling hydro-process plant. These considerations show that designing the back end of the fuel cycle from the beginning as a whole system is essential. When possible, closed fuel cycle offers many advantages from a sustainability point of view (waste minimization, preservation of resources, reuse of valuable materials, optimization of final disposal).

Paper ID#24 France (CEA), Molten Salt Reactor Technology – Mr Vincent Pascal

The experiences of operating Molten Salt Reactors (MSRs) are limited when compared to traditional solid fuel reactors (Light Water Reactors (LWRs), Sodium Fast Reactors (SFRs), etc.). MSRs have potential benefits, such as flexibility in terms of fuel isotopic enrichment, quick and efficient thermal feedback effects, a fuel loop without pressurization, natural convection capabilities, high efficiency of the energy conversion system. MSR development roadmaps need to resolve key issues including corrosion resistance of materials submitted to molten salts, the thermal load due to high temperature operation, the salt nuclear depletion and the fission products management, the handling of salt and the maintenance of components. French spent fuel reprocessing is based on the closed cycle strategy with the idea that fissile actinides in spent LWR fuel can be reused and valorised, which induces enriched uranium savings.

In 2020 CEA and ORANO launched preliminary studies around fast MSR technology in the framework of a common R&D effort for transuranic actinides’ management. Preliminary evaluations (at steady state) show a promising potential for the reduction of volume and long-term radiotoxicity of ultimate waste using MSRs as Pu+MA convertors, as a complementary service to Pu mono- and/or multi-recycling options. ISAC Project (CEA/CNRS/EDF/FRA/ORANO) aims to assess the potential of fast MSRs to enhance the French nuclear material management strategy.

Paper ID#25 France (Orano), Integration of MSRs in LWR-SMR Fleets to Close their Fuel Cycle and/or Manage Waste – Ms Isabelle Morlaes

Besides the standard options for the integration of LWR-SMRs in the current back end of the fuel cycle (i.e., direct disposal or multirecycling of plutonium), a preliminary study has been performed to assess the potential of fast chloride MSRs in a symbiotic fleet of SMRs to ‘burn’ Pu and MAs from the LWR-SMRs. Coupling the La Hague plant with fast MSRs converting actinides into fission products would add value to the standard LWR spent fuel reprocessing activity, in terms of natural resources and reduction of volume and long-term radiotoxicity of generated wastes. Most of these process steps are already available in La Hague at industrial scale to achieve this 'symbiotic' system. Using synergies with the industrial capabilities of La Hague can accelerate the development and deployment of such back end solutions for LWR-SMRs. It is recalled that the halide coming from the spent salt can also be recycled to recover this valuable constituent and to minimize the load of the waste with this element.

Paper ID#26 Canada (Moltex Energy), Application of a Graded Approach to the Concept of Fuel Recycling – Mr Olivier Gregoire

The Moltex Stable Salt Reactor-Wasteburner (SSR-W) represents a nuclear fuel cycle strategy that offers synergies with current fuel cycle and waste management options, with the prospect for long-term, sustainable energy production. Waste-to-Stable-Salt (WATSS) technology is proposed to fabricate fuel salt from spent fuel using pyrochemical process, keeping Pu with other actinides and some lanthanides. The presentation highlighted the lack of authoritative definitions for the concepts of "reprocessing" and "recycling" in some country's regulations and proposed an alternative definition for both concepts. During subsequent discussion it was noted that the recycling plant is foreseen to be associated to the SSR-W reactor developed by Moltex to form an overall package.

Paper ID#27 Finland (STUK and VTT), Finnish Perspectives on SMR Back End of the Fuel Cycle – Mr Ville Koskinen (STUK) and Mr Timothy Schatz (VTT)

The first part of the presentation covered the ongoing revision of Finnish energy legislation, which will take SMRs into account. Although no SMR projects are underway in Finland, interest has been expressed in their deployment for distributed local heating applications (Combined Heat and Power (CHP)), with LWR-SMRs currently being preferred. Initial regulatory review has indicated that storage and transport unlikely to present a challenge for current licensing processes, however the deployment of SMRs near urban areas may not be readily acceptable. Any new entity wishing to dispose of spent fuel will need to enter into commercial discussions with POSIVA for use of the limited unallocated capacity in the current Deep Geological Facility (DGF) or initiate their own DGF programme. Fundamentally the disposal concept is robust, but optimization of canister designs will be needed for shorter fuel. A nuclear safety and waste management R&D programme for the period 2023–2028 has recently been agreed in which SMR related activities are encouraged. Initial technical assessments of spent fuel characteristics from a reactor designed for local heating applications identified a lower inventory, heat and radiation arising from a lower target burnup but higher criticality hazard compared to a notional off the shelf LWR-SMR. The differences were considered insufficient to invalidate the current KBS3 disposal concept. However, the lack of fuel characterization, operational and maintenance experience and operating data limited the depth of analysis that could be undertaken. For significant deployment of small district heating units, centralized facilities for spent fuel and radioactive waste management would be attractive, however issues of ownership and liabilities would have to be worked out. Research has been started to understand whether the public would be accepting of SMRs being sited and waste management operations being undertaken in urban areas, with some mixed results.

Paper ID#32 Sweden (Studsvik), The Anita Programme and SMR Spent Fuel Management from the Swedish Perspective – Mr Kyle Jonson

ANITA is an academia-industry collaboration to achieve sustainable future to understand legal and technical issues to enable SMR deployment by the end of the decade. The programme will assess a wide range of legal and engineering aspects relevant to SMRs. Currently the spent fuel management levy in Sweden represents 25% of operating costs, i.e., a level that could affect competitiveness. Changes to fuel designs or characteristics needs to be approved by SKB, as SKB is responsible for storage and disposal of spent fuel in Sweden. The process is also applied to import of fuel for PIE. The extent and duration of the technical assessments reflect the degree of deviation from current technical options and needs to be backed by experimental data. Where required, any remediation steps need to be defined and liability agreed before acceptance. To underpin and support this process of acceptance, it needs data from irradiated specimens through test irradiation capacity and hot cell infrastructure for characterization and data generation.

Paper ID#33 Canada (CNL), Challenges of Small Modular Reactor Used Fuel Management in Canada – Mr George Xu and Mr Blair Bromley

A broad range of fuel types and SMR concepts are being evaluated, with work being prioritized by Canadian utility preferences and vendors potentially wishing to site demonstration of FOAK reactors in Canada. For remote locations the aim is to have very long batch fuel cycles. With HALEU options, previously developed schemes for fuel refabrication and re-irradiation in CANDU pressure tube heavy water reactors (PT-HWRs) could become more attractive because of higher residual enrichment and fissile content. Spent fuel inventories being assessed are based on open fuel cycle as a baseline consistent with current SFM strategy and planned infrastructure, pending completion of reprocessing studies and governmental reprocessing policy making. The range of assessed SNF heat generation rates from different concepts is very large. Pre-disposal management may be similar to that for current CANDU fuels, with remediation or processing and immobilization potentially being required for some options. In tandem with fuel recycling in the long term, transmutation has been looked at in terms of blanket assemblies in thermal-spectrum reactors, as a theoretical exercise to demonstrate that it is feasible, without any expectation that this strategy would be part of a preferred option that is ultimately implemented.

Paper ID#34 United States of America (DoE), Overview of US DoE's Office of Spent Fuel and Waste Disposition Activities – Mr Jorge Narvaez, Ms Natalia Saraeva and Mr Stephen Kung

Activities of DoE cover integrated waste management, R&D on waste management and international collaboration. On integrated waste management the current focus is on development and deployment of a consent-based siting process for centralized storage of current LWR fuels. Learning from this would ultimately also support SMR implementation activities. Waste management R&D is being undertaken to evaluate management and disposal options and inform decision making for current and SMR fuels, for which a range of examples were given.

Paper ID#35 United States of America (PNNL), Progress on Considering the Back End of the Fuel Cycle for Small Modular Reactors – Mr Stuart Arm

The focus of the presentation was on systems that could be adapted or deployed in the US in the near term. In relation to example decay heat generation rates at a common cooling time of 10 years, it was noted that these can vary substantially depending on management options selected for some fuels, specific examples given for molten salt and HTGR fuels. Whilst national regulations relating to storage can conceptually be applied to SMRs, there are many details to be worked through to address all nuances and to develop the data and designs required for licensing of such systems. Transport regulations were similarly generally applicable but there are significant gaps in understanding off normal transport conditions for licensing. Conditioning and treatment could be required for hazard reductions, regulatory compliance, economic viability or as a precursor to processing, depending on fuel type.

Work was underway to assess the conditioning potentially needed to transport and store MSR fuel materials. Interpretation of some regulatory terms, such as 'damaged fuel', would require further work, examples of MSR and HTGR fuels were again cited as examples with different characteristics. In summary, work is underway to understand the potential implications of SMRs' SNF in current LWRs' SNF management, and their possible post-discharge management options for subsequent disposal, which fell to DOE to resolve. Overall, work to date had recommended closer regulator-vendor interaction to develop and evolve regulatory requirements for SMR fuels.

Paper ID#36 United States of America (SNL), Investigations into Back End of the Nuclear Fuel Cycle (BENFC) Issues for Advanced Reactor (AR) Fuels and Accident Tolerant Fuels (ATF) – Mr Ramon Pulido

This presentation summarized current work on spent fuels for current and future reactors, including ATF and novel fuels. Characteristics of fuels that need to be understood to support assessments of disposal impact and hence inform integrated waste strategy for the back-end management of all prospective fuel types were identified. The identified characteristics are similar to the list shown earlier by Sweden. Preliminary work indicates that current approaches with engineering adaptions for different geometries are credits for ceramic fuels, but more substantial adaptions would be required for other fuels. Work is planned to examine ATFs and SMR fuels currently under irradiation testing once they reach full burnup. PIE work supporting back-end activities is based on that developed for the High Burnup Extended Storage Demonstration project 'sister rod' examinations.

Paper ID#37 United States of America (ANL), Evaluation of Advanced Reactor Spent Fuel Management Facility Deployment – Mr Milos Atz

This presentation showed results of work to map the key processes and milestones associated with delivering new spent fuel management infrastructure and the associated implementation risks. Key milestones included establishing responsibilities, siting, transport infrastructure, facility design, licensing and permitting, construction and commissioning. For each type of facility and each type of fuel, a qualitative assessment was made of the extent to which implementation would be more difficult than for current LWR fuels. This led to identify different challenges for different fuel types. The overall outcome led to identify commonality across many facilities and fuel types and identify recommendations for government activities for legal framework changes to facilitate deployment of required facilities.

Paper ID#38 United States of America (NEI), Opportunities to Optimize Small Modular Reactor (SMR) Used Nuclear Fuel Management in the United States – Mr Rod McCullum

The need to address new spent fuel management issues provides opportunities to drive improved solutions for fuel management. Over last few decades cost pressures have led to innovation in the development, deployment and optimization of dry fuel storage. Whilst the technical work required to develop and deploy management options for SMR fuels would require substantial innovation and development work, the industry has a good record and foundation for undertaking and delivering new solutions. Currently three suppliers — Holtec, Orano TN, and NAC International — are well positioned to meet US storage needs and continue to offer new and improved storage solutions in the quest to gain market share. They each possess highly sophisticated and proven analytical tools for evaluating every aspect of storage system performance (i.e., criticality, heat transfer, shielding, materials issues, etc.) that they can continue to refine and apply to the design of transportation and storage systems for SMR used fuels. There is a re-emergence of recycling as part of the US fuel cycle that will create opportunities to produce tailored waste forms designed with long-term storage, transportation, and disposal in mind. SMRs deployed along with recycling technologies have the potential to redefine the very nature of used nuclear fuel, both for that discharged from SMRs as well as much of the existing inventory.

Paper ID#41 Romania (NRWA), Romania Strategy for Radioactive Waste and Spent Nuclear Fuel according with the Development of the Expending Nuclear Programme – Ms Alice Mariana Dima

Currently, Romania has decided on an open fuel cycle approach, considering SNF as high-level waste, which is to be disposed of in a deep geological repository. The CERNAVODA NPP has two CANDU-6 reactors (each 700 MW), wet storage in the spent fuel bay for a minimum of six years cooling and dry storage in the dry spent fuel storage facility (DICA) for 50 years. Romania also has Research Reactors: a 14 MW TRIGA reactor and a WWER research reactor that was shut down in 1997. The National Strategy establishes two main solutions for the safe management of radioactive waste and spent nuclear fuel: implementing a near surface repository for low and intermediate level, short lived waste (DFDSMA); and a Deep Geological Repository for long lived radioactive waste & spent nuclear fuel. The new nuclear installations planned in Romania are the Advanced Lead Fast Reactor European Demonstrator (ALFRED) and the Extreme Light Infrastructure-Nuclear Physics (ELI-NP). Romania is also pursuing the implementation of NuScale SMRs, with an initial project for implementing 6-module, 462MWe NPP of 77 megawatts each, with the possibility of expanding the number of modules and/or to other sites. An updating national strategy for safe management of SNF and RW in Romania will need to take into consideration an economically feasible solutions for integrating CANDU fuel cycles with new generation plants considering the implementation of DUPIC cycle in Romania. As many other countries with relatively recent nuclear power programmes, the geological repository programme in Romania is at an early stage and takes into consideration all technical solutions for disposal of the SNF and HLW.

Paper ID#42 France (Orano), Challenges for Transport and Storage – Mr Brut Stephane

Storage and transportation are critical parts of the overall management of the front end and back end of the fuel cycle. In addition, licensed solutions need to be available for the various fuel types to be transported. Anticipating the package solutions is key to optimized transport scenarios with integration early in the reactor design phase. The national regulations based on the IAEA Safety Standards, and the radioactive material characteristics (radioactivity, fissile material) defines the type of package, the safety requirements, and tests to be conducted. The design features for fissile material including an assumption of water ingress inside the package has to be considered in the criticality analysis, unless the package is double barrier design. The package design for fresh fuel with a high fissile content has to be considered including all by-products resulting from the manufacturing chain. For example, UF_6 transport with enrichment higher than 5%. The simple barrier design with high fissile content subcriticality may reduce the payload capacity. Double barrier design enhanced reduction of package numbers and transport cost and the complexity of the design and additional operations. For spent fuel transport and storage, heavy cask due to shielding requirements with high density materials, large cask with impact limiters for transport, and compatibility with the loading facility process ought to be considered. Therefore, the existing solutions like TN-EAGLE and NUHOMS EOS and systems, licensed for LWR fuel, can be adapted/modified for such needs.

Paper ID#43 UK (NTS), Transport Considerations for SMR Fuel Cycle and TNPPs – Mr George Burnett

The UK is planning to seek approval of one new reactor per year until 2030. UK SMR is expected early 2030s and potentially more licensed sites across the UK, as UK SMR is looking to deploy on previously licensed sites. In the IAEA Booklet on Advances in Small Modular Reactor Technology Developments, examples of nuclear fuel transport experiences can be identified, such as Energy Well, MicroURANUS, ELENA and eVinci. The Transport Security Approach for SMR Applicability implies: Categorise material (type, form, and quantity) adopting a graded approach; Sabotage considerations (especially for back end); Incorporate Design Basis Threat (DBT) / Threat Assessment; Implement defence in depth and remote transport and siting. The Transport Gaps can be summarized as: Transport safety substantiation of fuel characteristics (SSR-6 Normal Conditions of Transport and Accident Conditions of Transport); Data availability for Package Design Safety Report requirements and Package availability unknown; and Inherent security characteristics (theft vs sabotage). Some insights on safety and security aspects for transportable nuclear power plants were highlighted.

Paper ID#44 Argentina (CNEA), Back End the Fuel Cycle Considerations for CAREM the Argentinian Small Modular Reactor – Ms Laura Kniznik

CAREM-25 design is based on an integrated LWR, using enriched uranium as fuel. It is an indirect cycle reactor conceptually simple, which offers a high safety level. The CAREM fuel elements are hexagonal section with 127 rods, of which 108 are fuel rods, 18 are guide tubes for absorbing elements, and one is an instrumentation tube. The reactor core has 61 fuel elements. The qualification of fabrication process for enriched uranium fuel pellets began towards the end of 2018. The foreseen options for managing CAREM’s spent fuel are similar to the approach of other NPPs in Argentina: wet storage to allow sufficient decay of the fission products and later dry storage on the reactor site. Under normal conditions, radioactive waste generated by CAREM-25 will be low or intermediate level. The CAREM-25 design provides for radioactive waste storage within the CAREM site.

Paper ID#45 UK (NNL), Managing Fuel from SMRs: UK Framework and HTGR Gap Analysis – Mr David Hambley

The UK remains committed to nuclear power as a key technology for meeting net zero goals by 2050. New large LWRs are under construction and planned to replace existing reactors that are expected to cease generation by the end of this decade. The UK is supporting development of a range of SMR and advanced nuclear technologies, with anticipated implementation starting in the early 2030s. The UK is now operating an open fuel cycle, although reprocessing remains an option if economically and environmentally attractive. Management of current spent fuels is mature and consistent with national strategy. The siting process for a deep geological disposal facility is underway with four candidate community partnerships having been established. The UK regulation provides mechanisms that ensure appropriate consideration of the whole fuel cycle through development, licensing, operation and modification of reactors and fuels.

For the HTGR, the technological options for spent fuel management are non-dismantling (co-disposal of fuel components and graphite materials) and dismantling (core components separation). Current priorities are associated with determining the conditions under which several degradation phenomena may be of concern, principally: longevity of TRISO containment layers in groundwaters, oxidation of TRISO containment layers and fuel graphite components, effects of He pressurisation on the increase of failed fuel proportion during timescales relevant for long-term storage and disposal, and the extent of fission product migration into and through TRISO containment layers.

Paper ID#46 Poland (INCT), TRISO Fuel Management Depending on the Choice of the Fuel Cycle Research Currently Conducted at INCT, Warsaw, Poland – Ms Katarzyna Kiegiel

Poland plans to have nuclear power from about 2033 including small modular reactors based on high-temperature reactor technology. Construction of the first HTR reactor with a capacity of approximately 150– 300 MW is planned before 2031. Open and closed fuel cycle options are being considered for the resulting TRISO spent fuel: an open fuel cycle is currently proposed, although a closed fuel cycle may be considered in the future. At present, the onsite storage of spent nuclear fuel from Polish nuclear reactors is recommended, from where it can be transferred to the deep disposal facility, when available or reprocessed. High-level waste from reprocessing will be disposed of in this deep repository, too. Institute of Nuclear Chemistry and Technology is involved in European projects on radioactive waste management and has experience in studies on the safe management of radioactive waste from the fuel cycle. The numerous studies being carried out concentrate on the development of the procedures suitable for the HTR waste management. It seems that improved extraction methods based on the currently employed ones for reprocessing spent nuclear fuel will be suitable for managing spent fuel from Gen-IV reactors.

Paper ID#47 Japan (JAEA), Fuel Cycle Scenarios and Back End Technologies of HTGR in Japan – Mr Yuji Fukaya

JAEA introduced the status of R&D on back end technologies for HTGR fuels and technological subjects to improve the specifications for some fuel cycle scenarios. Japan had developed reprocessing technologies for LWR spent fuels based on French technologies. R&D on back end technologies for HTGR spent fuels is necessary for demonstration at industrial level. HTGR technologies had been developed with the assumption of reprocessing in Japan. The study of the head-end process for HTGR spent fuel reprocessing has been completed, and its applicability in Rokkasho Reprocessing Plant (RRP) has been confirmed. The disposal technologies for vitrified waste apply to HTGRs, and feasibility for direct disposal of HTGR spent fuels had been also confirmed.

A near field model for graphite waste dose evaluation could be developed to improve the specifications for fuel cycle options. The waste may be disposed of more easily from shallow-ground pit disposal to shallow-ground trench disposal. The recovery ratio of reprocessing with HTGR head-end process needs to be confirmed to achieve potential toxicity reduction for multi-recycling option.

Paper ID#48 China (Tsinghua University), Deconsolidation of HTR-10 Irradiated Fuels & Measurement of Burnup for SFE Storage – Ms Xiaotong Chen

The commercialization of HTGRs in China highlights the storage challenge of spherical fuel elements (SFEs). The burnup measurement of SFE is an important issue. Three SFEs with low to medium burnup were selected and measured using destructive and non-destructive methods. Gamma and mass spectrometry were applied, and an electrochemical deconsolidation process was conducted to obtain TRISO fuel particles from specific regions. The uniformity of burnup in each SFE was also studied simultaneously by a mass spectrometry method. Gamma spectrum of three SFEs (P1, P2, P3) were first collected by an online burnup measurement system (BUMS) before unloaded from HTR-10 and transferred to the hot cell, where provided a much lower radiation background and longer live time than at the reactor. Radioactivity of caesium fission product was sophisticatedly measured. The main radionuclides like U-235, U-238, Cs-137 and Nd-148 were measured by radiometric and mass spectrometric methods. The fractional U-235 burnup was used to determine the burnup distribution, which proved that in each SFE, the burnup is relatively uniform. The measured burnup using gamma spectrometry by Cs-137 compared well with the one using mass spectrometry by uranium nuclides.

3. SUMMARY OF BREAK OUT SESSIONS

Prepared by
Mr Jorge Narvaez (United States of America, Office of Nuclear Energy (DoE))
Mr Timothy Schatz (Finland, Technical Research Centre of Finland)
Mr David Hambley (UK, National Nuclear Laboratory)
Mr Andrew Worrall (United States of America, Oak Ridge National Laboratory)

Three breakout sessions were conducted to identify and discuss gaps/challenges/opportunities for implementing back end of the fuel cycle strategies for different SMR Technologies (LWRs, HTGRs, and Advanced Reactors including MSRs) as well as to discuss integration of different SMRs' fuel cycles with the ones already implemented for the current fleet of LWRs, focusing on the management of spent fuel.

The key points raised in each of the sessions by the meeting participants have been reproduced.

3.1. LIGHT WATER REACTOR TYPE

Fuel Type

There is uncertainty related to accident tolerant fuel (ATF) and High-Assay Low-Enriched Uranium (HALEU) fuel availability.

Fuel Data

It is important to characterize the spent nuclear fuel (SNF) that will be produced from this type of SMRs. This data affects transportation-related activities.

Transportation

A question was raised whether existing cask designs could be used to transport SNF from SMRs, or design and licensing of new casks for this specific purpose would be required.

SMRs and their associated SNF and other waste forms may need to be transported back to their countries or points of origin. International land and maritime considerations need to be taken into account to properly plan for the future transportation of SMRs and associated waste. These considerations can also apply to the other type of envisaged SMR technologies.

Decommissioning

From the early stages of the design phase, reactor decommissioning is an important consideration. This presents a challenge as well as an opportunity to engage communities in decommissioning-related activities.

Reprocessing

For LWR-SMRs that are expected to use HALEU fuel, reprocessing could be a viable option.

Disposal

There is an opportunity to engage with nuclear embarking countries on the feasibility of using deep borehole disposal (DBD) as an alternative to DGRs for spent nuclear fuel disposal.

Fuel Cycle Back End

There is an opportunity to encourage SMR and Advanced Reactor vendors to collaborate with Member States on how to successfully navigate the back end of the fuel cycle.

3.2. HIGH TEMPERATURE GAS-COOLED REACTOR TYPE

Fuel Type

HTGRs use TRISO fuel that is typically fabricated into one of two forms: spherical pebbles or cylindrical blocks.

Choosing one option versus the other will likely come down to the country preference on refuelling type (online for pebbles and shutdown for blocks) and availability of the containers that will contain the irradiated waste forms.

Damaged Fuel

There is a need to characterize and define what 'damaged fuel' looks like for TRISO fuel.

A potential challenge deals with the identification of damaged fuel during normal operation and the mechanisms needed to remove and manage the failed fuel. Whether damaged fuel characterization is done at the TRISO particle level, at the encapsulation level, or at both is an issue to be resolved.

Fuel Data

Data is needed to support the characterization of the fuel and the expected waste forms. This will likely help expedite the licensing process. It is also important to obtain criticality data as the TRISO particles are likely to be fabricated using HALEU fuel.

Transportation

It is important to account for accident scenarios during the transportation of TRISO spent fuel and associated wastes.

Storage

There are decades-long experience in air storage of TRISO spent fuel. In contrast, there is little to no experience in wet storage; thus, wet storage is not considered a viable option for TRISO spent fuel.

Reprocessing

Reprocessing of TRISO fuel is thought to be possible and only small-scale research has been conducted to date. Nevertheless, any processes will likely need to plan for how to best deal with the release of C-14. The cost-effectiveness of reprocessing needs to be accounted for when determining its feasibility. One key challenge is determining whether the reprocessed fuel can be recycled into fresh fuel for another reactor. Finding reprocessing alternatives will likely be advantageous.

Disposal

TRISO spent fuels from several experimental reactors are currently being used to investigate disposal alternatives. However, more research and development are needed to conduct direct disposal of TRISO spent fuel. The consensus is that TRISO fuel is thought to be disposable after irradiation.

There is a need to identify the infrastructure framework (primarily for transportation) required for potential future international disposal.

Energy Mix

There is the potential for the integration of HTGR SMRs with hydrogen production or process heat. One key challenge is to determine the back-end factors that could make this integration possible.

Safeguards

Some countries consider that HTGRs are an unattractive option to be deployed due to the material accountancy issues that arise when dealing with the hundreds or thousands of pebbles required for normal reactor operation which is a challenge from a safeguards perspective.

3.3. ADVANCED REACTOR AND MOLTEN SALT REACTOR TYPE

Fuel Type

Most advanced reactor designs include non-LWR fuel types. Some representative fuels are metallic, oxide, nitride/carbide, TRISO, molten salt, etc.

To understand the SNF, it is very important to first characterize the fresh fuel types proposed from the reactor vendors.

The fresh/SNF can be characterized by some of the following factors: chemical and physical properties, isotopic composition, neutron and gamma spectra, heat, burnup, cooling time, corrosion products, etc.

However, there are several challenges and gaps due to the diverse set of fuel types.

- To understand each fuel type, it is necessary to have cooperation from the vendors. This could be challenging due to intellectual property considerations;
- There is a need to create and maintain databases that could help researchers simulate the operation of these reactors. This could help to inform back-end activities such as storage, transportation, and disposal.

SNF 'Take Back' Option

The option of 'take back' is being discussed and poses several challenges. This option occurs when a vendor or country agrees to take back the SNF or parts of the reactor (entire reactor vessel for microreactors) from another country after reactor shutdown. This option can also apply to the other SMR envisaged technologies.

International agreements regarding SNF take back could be considered and put in place in anticipation of SMR and microreactor deployment.

Transportation and Storage

Packages that can accommodate the different fuel forms need to be designed and qualified.

One key challenge is to determine if one type of package can accommodate different fuel forms.

For molten salt reactors, an important challenge is determining if the SNF from these reactors will be in a solid or liquid form. It is important to identify if the expected form will remain the same under postulated accident conditions.

4. SUMMARY OF GENERAL DISCUSSIONS

Prepared by Mr Jorge Narvaez (United States of America, Office of Nuclear Energy (DoE))

A general discussion took place after the presentation of the findings during the three break out sessions mentioned before. This covered cross-cutting challenges, issues, and opportunities.

Safeguards

A discussion on how to work with the vendors about making their designs proliferation resistant took place.

Some general insights were provided by the IAEA Safeguards staff:

- Proliferation resistance is analysed starting from the preliminary design and reassessed as the design evolves.
- Guidelines are in place to ensure that nuclear material that enters a facility is located where it is supposed to be, and the design of a facility will not produce other undeclared nuclear materials.
- Several technical scenarios need to be considered. For example, a scenario in which a reactor operator or Member State use a facility to divert or produce other material. The more scenarios, the more complex the safeguards approach becomes.
- A facility's inherent proliferation resistance is a consideration in the introduction of safeguards for the facility and can potentially simplify these. However, some proliferation-resistance measures (e.g., plutonium isotopic assay in spent fuel) will not affect safeguards, and some proliferation-resistance measures can inhibit the application of safeguards (noting that safeguards will always be required as they represent independent verification under a State's safeguards agreement with the IAEA). The degree of 'safeguardability' in facility design is an important component of proliferation resistance that benefits from early consultation with the IAEA.
- Regarding encapsulated or sealed reactor modules, there is currently no specific safeguards approach as this will need to be country-specific and built around the vendor's design. The use of cameras and other unattended monitoring equipment is considered for places where direct human verification is not possible or inefficient.
- It is unknown at this time how a State's Comprehensive Safeguards Agreement (CSA) with the IAEA (particularly the relevant subsidiary documents detailing implementation) will need to be amended to allow for the deployment of SMRs, as this depends on the safeguards approach developed once detailed design information is declared to the IAEA.

Other Topics

Fuel characterization is needed to anticipate what to do with the SNF. In most of the countries, a decommissioning plan is required prior to the operation of any nuclear power plant.

There was a proposal to review the definitions of reprocessing and recycling in the next revision of the 'IAEA Safeguards Glossary'.

Several participants expressed interest in the addition of the terms ‘social acceptance’, and ‘public engagement’ when involving members of the public in the activities related to the back end of the fuel cycle.

Coordinated Research Projects (CRP)

Here are some ideas of projects that some of the meeting participants expressed interest in seeing explored:

- Characterization of the fuel, spent nuclear fuel, and other waste forms associated with the operation of advanced reactors and SMRs;
    - This is something that could be done in parallel with the IAEA SMR Booklet;
    - It could be added to the current IAEA ARIS database as additional information on the back end of the fuel cycle from different SMR designs;
- Several participants expressed interest in the infrastructure needed to deploy advanced reactors and SMRs, and the cost analysis associated with this;
    - The ‘human infrastructure’ needed to license and operate these facilities could also be considered. The number of jobs and the career types need to be identified as well;
- Transportation security is an important consideration for these types of reactors, as well as for the regulations needed to transport HALEU fuel;
- Develop a simulation tool for advanced reactor fuel cycles;
    - The current IAEA simulation tool, the Nuclear Fuel Cycle Simulation System (NFCSS), is currently only capable of simulating thermal reactors as PWRS, BWRs, PHWRs, RMBKs, AGRs, GCRs, WWERs (UOX, MOX and ThOX fuel cycles). However, work is in progress to simulate advanced reactor fuel cycles;
    - New modules will be added to the simulation tool to address the footprint for disposal of advanced reactor fuel cycles.

5. CONCLUSIONS AND FUTURE AREAS OF WORK

This chapter reflects the main conclusions of the Technical Meeting and the recommendations of participants to the IAEA for future activities in the field of the management of spent fuel from the different SMR technologies.

Prepared by Ms Cécile Evans (France, Orano)

For countries embarking or willing to embark on an SMR programme, whether they are nuclear countries or newcomers, understanding the implications of the spent fuel management programme that would need to be undertaken is important to make informed decision on the specificities of different SMR technologies and on the fuel cycle options.

For various technologies/families of technologies, describing activities to be developed and implemented to manage spent fuel up to the disposal of HLW will enable identification of:

- The various steps to be undertaken, their timeline and duration.
- The data required to develop the various fuel cycle options, to predesign the back-end programme based on collected data on mass flows of materials and wastes; isotopic, chemical form, impurities, waste forms and their compatibility with disposal; building on from existing knowledge/boundaries conditions acquired so far and enabling sharing of information.
- The data to be collected from irradiated fuel and their use in designing systems for licensing.
- The gaps with existing practices/technologies/infrastructures developed for existing systems and the specific characteristics associated to SMR deployment as well as the opportunities to develop new technologies to fill the gaps.
- Which infrastructures, including their size, would need to be developed, whether they would be locally implemented or based on existing industry solutions/services, including cost elements.

This would require establishing specific roadmaps of activities to be developed per technology, identifying what can be derived from existing practices, optimized, adapted, or fully developed considering the lack of data, gaps with existing knowledge, and defining required additional data and the way to acquire them.

This would allow for comparison of various reactor technology systems, comparing fuel cycle options to identify/quantify the effort required to implement a spent fuel management strategy in terms of nuclear facilities, technology developments, types of nuclear materials involved, generated radioactive waste forms and other infrastructures needed such as human resources, regulatory framework, financing, etc. In addition, the need to develop/reach public engagement merits specific emphasis.

This work will identify and highlight key parameters for designing the back-end programme of the different fuel cycle options associated with the different SMR technologies.

These roadmaps could be developed by the IAEA in the framework of a Coordinated Research Project (CRP) with the main objectives of:

— Identifying viable nuclear fuel cycle options for the different SMR technologies;
— Establishing generic key parameters that would then allow a country to develop from that tool their analysis incorporating their specific context;
— Identifying common technologies/similarities for various reactor types and/or significant differences.

Thus, there would be merits in having one CRP addressing the back end of various SMR technologies, to ensure that synergies and cross-cutting issues will be identified.

LIST OF ABBREVIATIONS

ALFRED	Advanced Lead Fast Reactor European Demonstrator
ARIS	Advances in Small Modular Reactor Technology Developments
ATF	Accident Tolerant Fuel
BENFC	Back End of the Nuclear Fuel Cycle
BUMS	Online Burnup Measurement System
CHP	Combined Heat and Power
CRP	Coordinated Research Project
CSA	Comprehensive Safeguards Agreement
DBD	Deep Borehole Disposal
DBT	Design Basis Threat
DGF	Deep Geological Facility
DFDSMA	Final Disposal Facility for Radioactive Wastes
DICA	Intermediate Dry Spent Fuel Storage Facility
ELI-NP	Extreme Light Infrastructure-Nuclear Physics
EPZ	Emergency Planning Zones
FOAK	First of a Kind
HALEU	High-Assay Low-Enriched Uranium
HEU	Highly Enriched Uranium
HLW	High-level Waste
HTGR	High Temperature Gas-Cooled Reactor
HTR	High Temperature Engineering Test
ISAC	Innovative System for Actinides Conversion
LEU	Low Enriched Uranium
LWR-SMR	Light Water Reactor-Small Modular Reactor
LWR	Light Water Reactor

MA	Minor Actinides
MOX	Mixed oxide fuel
MSR	Molten Salt Reactor
NFCSS	Nuclear Fuel Cycle Simulation System
NPP	Nuclear Power Plant
PASSED	Performance Assessment of Storage Systems for Extended Durations
PIE	Post-irradiation Examinations
PT-HWR	pressure tube heavy water reactors
PUREX	Plutonium Uranium Redox Extraction
PWR	Pressurized Water Reactor
RRP	Rokkasho Reprocessing Plant
RW	Radioactive Waste
SFE	Spherical Fuel Elements
SFERA	Spent Fuel Research and Assessment
SFM	Spent Fuel Management
SNF	Spent Nuclear Fuel
SPAR	Spent Fuel Performance Assessment and Research
TNPP	Transportation Nuclear Power Plant
TRISO	Tri-structural Isotropic
WATSS	Waste to Stable Salt
WCR	Water-Cooled Nuclear Reactor

EXTENDED ABSTRACTS/FULL PAPERS SUBMITTED AND PRESENTED AT THE TECHNICAL MEETING

Paper ID#12

JRC CONTRIBUTIONS TO SMR AND THE BACK-END FUEL CYCLE

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Abstract

Part of the JRC mandate is to implement the EURATOM research and training programme, which is a part of the EU's efforts to further develop technological leadership and promote excellence in nuclear research and innovation. In the current European energy mix, nuclear power plants are mainly used to generate electricity. However, the nuclear sector is developing towards the production of electricity integrated in an energy system with high share of renewable energy sources (RES), as well as toward the production of heat to be delivered to high energy intensive industrial sectors and districting heating. The SMRs raise high expectations in many areas and also challenges and hurdles that the nuclear sector needs to address to deploy them, for instance the waste management where an integration with existing back-end processes and schemes is desirable but needs to be verified.

1. INTRODUCTION

As the science and knowledge service of the European Commission (EC), the JRC mission is to support European Union (EU) policies with independent evidence throughout the whole policy cycle. The JRC is dealing with growth and innovation; energy, transport and climate; sustainable resources; space, security and migration; health, consumers and reference materials; nuclear safety and security as well as with knowledge management and competences [1]. Part of the JRC mandate is to implement the EURATOM research and training programme, which is a part of the EU's efforts to further develop technological leadership and promote excellence in nuclear research and innovation, in particular ensuring the highest standards of safety, security, safeguards, radiation protection, including safe spent fuel / radioactive waste management and decommissioning whereby contributing to social well-being, economic prosperity and environmental sustainability. The general objective of the 2021-2025 EURATOM Programme is to pursue nuclear research and training activities, with an emphasis on the continuous improvement of nuclear safety, security and radiation protection, as well as to complement the achievement of Horizon Europe's objectives inter alia in the context of the energy transition [2].

The EC proposes and supports since long-time political and technological frameworks for the mitigation of climate change and air pollution in the EU, putting attention to several aspects as for instance electricity, heat, transport, to name but a few. In this framework, the EC has started several activities to boost collaboration among the EU Member States for achieving common energy and climate targets. In the energy sector noticeable initiatives have been the Strategic Energy Technology Plan (SET-Plan) launched in 2008, the Energy Union Strategy of 2015 and the European Green Deal as published in December 2019 [3].

In its communication “A Clean Planet for all. A European strategic long-term vision for a prosperous, modern, competitive and climate neutral economy” the EC has provided various scenarios for EU-27 in order to achieve the Paris agreement climate goals of keeping global warming well below 2°C and pursuing efforts to limit the increase of temperature to 1.5°C by the year 2050. All scenarios assessed in this study foresees a share of nuclear energy that varies between 10 % and 15 % [4].

2. DISCUSSION

In the current European energy mix, nuclear power plants are mainly used to generate electricity, with some exceptions. However, the nuclear sector is developing towards the production of electricity integrated in an energy system with high share of renewable energy sources (RES), as well as toward the production of heat to be delivered to high energy intensive industrial sectors and districting heating. This development points on Small Modular Reactors (SMR) and several EU Member States have shown an increased interest in them. The SMRs raise high expectations in many areas, e.g.:

- Lower fabrication costs and reduced construction time due to simplified design, serial factory fabrication of modules and self-financing scheme;
- Pre-Licensing/Licensing Harmonization (in EU and internationally);
- Flexibility and load-follow capability allowing for integration with intermittency of RES;
- Significant safety advantages (relevant also for public acceptance);
- Offer electrical output at lower cost per kWh;
- Replacement of fossil fuel plants for heat generation and industrial processes/factories.

Along with this expectation there are also challenges and hurdles that the nuclear sector needs to address to deploy SMRs, for instance the waste management where an integration with existing back-end processes and schemes is desirable but needs to be verified. The many SMRs concepts under development [5] show that there are various fuel compounds and configurations proposed as e.g., high assay low enriched uranium (HALEU), oxides or nitrides fuels, TRISO particles, liquid fuel envisaged for irradiation in thermal or fast neutron spectrum. Moreover, higher fuel burnups and different fuel cladding materials are proposed as well. Finally, the different waste forms arising from an open or a closed fuel cycle would be an additional factor impacting the waste management strategy.

In order to anticipate potential SF management and disposal solutions to be implemented for SMRs deployment, the JRC is performing preliminary studies using spent nuclear fuel (SF) from current programmes as analogue. In general, the objective of these studies is to assess the ability of the spent fuel / waste form to fulfil its expected function over long time periods. In particular, the JRC focusses on providing experimental data for the development and validation of physical models and numerical codes aimed at assessing safety aspects of the SF long term properties and behaviour. The studies are in the area of:

- Extended Interim Storage: Ageing processes occurring during storage of SF rods are safety-relevant both for storage and for handling/transportation procedures foreseen at the end of the storage period. Mechanisms that may cause severe degradation of the mechanical integrity of the SF rods are dependent on the radioactive decay power generated in spent fuel, the behaviour of hydrogen present in the metallic cladding of the fuel rods, and the build-up of alpha-decay damage and helium affecting the mechanical properties of the spent fuel [6];

- Accidental Conditions: SF rods are handled in pools, dried, transported, wet- or dry-stored and retrieved for further management/disposal thereafter and during these operations potential accidents might occur. The consequences of these accidents causing spent fuel rod failure may involve fuel particles release and dispersion. The failure mode of the SF rod is related to the aging mechanisms as mentioned in the previous point. As shown in Figure 1, mechanical testing allows the assessment of the amount and size of particle released in such extreme cases for safety assessment and, if needed, development of a remediation strategy [7].

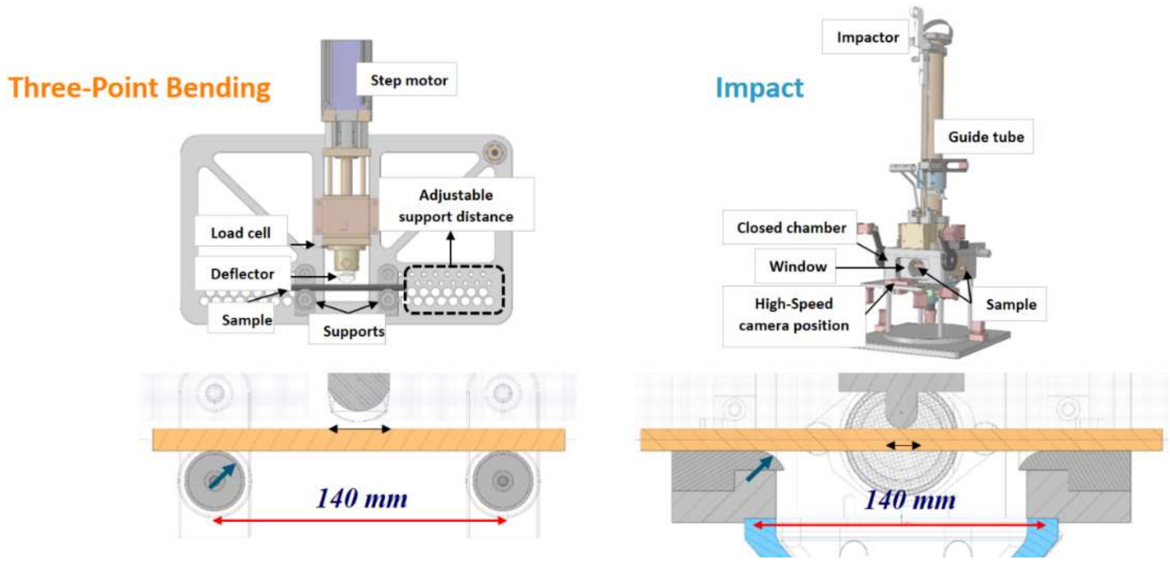

FIG. 1. Schematic view of bending and impact tests used at JRC for mechanical testing of SF segments. The results of the tests performed so far indicate that the same order of magnitude of fuel mass is released from UO_2 and MOX over a wide range of burnup (reproduced from Ref. [7] with permission).

— Geological disposal: The main objective of the studies is to provide experimental data needed to reduce uncertainties on release of long-lived radionuclides over an open-ended disposal time scale. The data are mainly related to the effect of the environment on the matrix corrosion and the associated instant release fraction and source term [8]. Parameters that affect the long-term stability of the spent nuclear fuel are related to both the spent nuclear fuel conditions (e.g., radiation history and the repository conditions as shown in Figure 2.

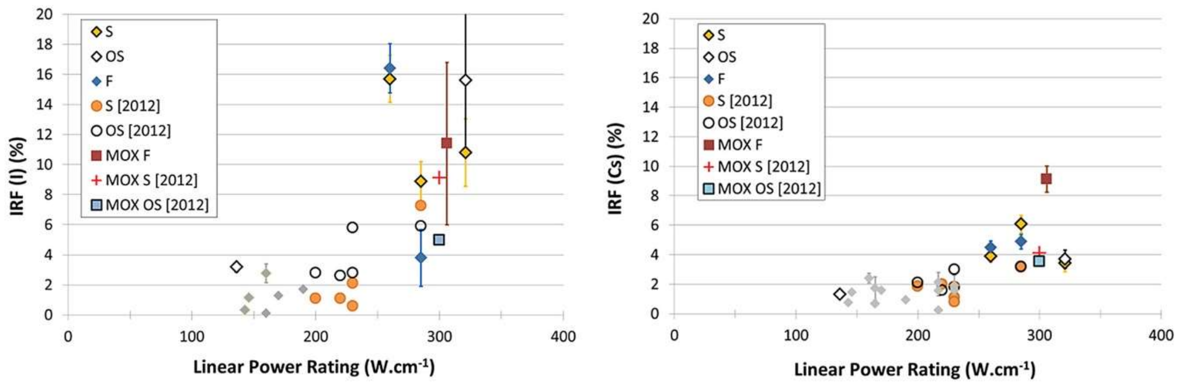

FIG. 2. Correlation between Instant Release Fraction (IRF) and irradiation history. The linear power rating is the main factor affecting the IRF (reproduced from Ref. [9] with permission).

In Summary, there is an increased interest in several EU Member States towards SMRs. The interest for these reactors is to replace GHG emitting power plants and/or to integrate with Renewable Energy Sources, for electricity production as well as heat generation for industrial use or district heating. There are several SMR designs under development with different coolant technologies, fuel forms and back-end fuel cycle. Verifying the compatibility of the SMRs waste with the existing waste management schemes is important to identify specific aspects that could possibly affect the back end of the SMR fuel cycle. The JRC has started to perform this assessment working with analogues and leveraging expertise and capabilities acquired on LWR spent fuel / nuclear waste.

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Paper ID#14

THE POTENTIAL IMPACTS OF SMRS ON MULTINATIONAL COOPERATION AT THE BACK-END OF THE FUEL CYCLE

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Extended Abstract

For all countries operating or considering utilization of nuclear power, safe and affordable spent fuel management strategies need to be a key goal. Currently, there is great interest in the potential for SMRs. A wide range of prototype designs is being pursued by both national development agencies and commercial vendors and the global interest includes countries with both small and large nuclear power programmes. The potential wide adoption of SMRs raises novel issues related to the back end of the fuel cycle and some of these impacts directly on prospects for enhanced multinational cooperation on the management and the disposal of the radioactive wastes from SMRs.

Any programme adopting SMR technologies will need to consider the implications for national radioactive waste management (RWM) policy and strategy and for the national waste inventory. Existing policies and plans might need to be revisited to take account of the different types of waste materials generated by SMRs. For example, the range of SMR concepts can use significantly different fuel matrices and assemblies in different configurations to those that might be being managed from conventional reactor types. Technical issues that are likely to need consideration include: integrating SMRs into national fuel-cycle arrangements, reprocessing and storage policy, packaging SMR wastes for disposal, choosing most appropriate deep geological repository (DGR) concepts and designs for SMR fuel and HLW, and assessing NPP end-state impacts on decommissioning and RWM policy. These technical factors form some of the boundary conditions that will define how a country considering adoption of SMR technologies will interact with other countries or groups of countries in terms of sourcing SMR technology and managing radioactive wastes.

Consideration of these multinational aspects forms the core of an on-going study being supported by the USDOE and the ERDO Association. The commercial and strategic issues of an international nature that are being evaluated, along with the associated national and international policy issues, include the selection of potential supplier countries of SMR technologies, fuels and materials, the possibility of fuel/waste take-back arrangements with technology and/or fuel suppliers and the impacts on arrangements being made for involvement in multinational RWM solutions, including multinational repository (MNR) projects. The impacts on MNR projects are of importance not only for countries that might deploy SMRs but also for small inventory non-nuclear power nations that would benefits from the existence of an MNR.

The project is divided into two stages, each with several tasks. The preliminary stage 1 covers technical matters and sets the boundary conditions for stage 2, which is the main focus of the project and covers the international policy and strategy issues outlined above.

The technical issues that will be explored are broken down into a series of covering the following topics:

— SMR technologies and suppliers: a high-level tabulation focused on the current and likely near-future deployable SMR technologies;

— SMR fuel characteristics: a high-level evaluation of the characteristics of the spent fuel produced by the numerous designs which have been proposed for SMRs;
— SMR fuel disposability: an outline comparison of SMR fuel characteristics relevant to their disposability;
— SMR operational and decommissioning wastes: a scoping assessment of the types and amounts of fuel and other wastes generated during the lifetimes of the different SMR technologies;
— SMR impacts on management of a national nuclear fleet

The information base and scenarios generated in stage 1 will be used as the basis for an assessment of how SMR deployment might affect the international fuel cycle regime in general and, more specifically, multinational initiatives for RWM. Topics to be covered are:

— Strategic aspects of the international SMR market: an evaluation of the potential supply and demand landscape;
— Costs of SMR fuel waste management: a preliminary evaluation of the likely cost implications of disposal of SMR fuels;
— Impact of SMRs on MNR planning assessment of how a shared or a commercial MNR project could be impacted in terms of concept/design, economics, and scheduling if a number of users were to require disposal of SMR fuels and wastes.

Paper ID#16-1

SAFETY IMPLICATIONS ON THE BACK END OF THE FUEL CYCLE FOR SMRS

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Extended Abstract

Small modular reactors could offer substantial advantages relative to conventional LWRs, such as advanced safety features which relay only on passive safety systems, and enhanced flexibility in deployment and operation. However, the back-end part could pose challenges that should be addressed in a timely manner.

In this paper the impact of design changes on spent fuel characteristics relevant to the safety of the back of fuel cycle are presented. Particularly, the effects of increased neutron leakage due to small core size, core heterogeneity as well as increased fuel enrichment proposed in several designs of SMR are considered and their impact on safety is discussed.

Particularly, increased burnup could lead to a higher discharge burnup therefore accumulation of a larger quantity of transuranic elements. This influences decay heat and gamma/neutron dose potentially leading to longer cooling times in spent fuel pools or design modifications of already deployed spent fuel storages and spent fuel reprocessing facilities. Also, criticality safety analysis crediting fuel burnup will be affected.

In some LWR, SMRs boron is not used for reactivity control which leads use of the increased amount of control rods that could lead to the more heterogeneous distribution of important isotopes relevant to criticality safety with burnup credit and in some extent to decay heat and dose rates.

Paper ID#16-2

IMPORTANT ASPECTS OF FUEL CYCLE TO BE CONSIDERED DURING DEPLOYMENT OF NEW NUCLEAR POWER TECHNOLOGIES IN ARMENIA

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Abstract

As Armenia considers implementation of SMRs, as one of the options for future electricity generation installations, considerations of fuel cycles for SMRs is important since it can create challenges in spent fuel and radioactive waste management. Management of spent nuclear fuel is one of the most important issues to be solved during the planning of deployment of new nuclear technologies in Armenia.

1. INTRODUCTION

According to the country's current strategic plan for energy sector development for 2015-2036, a nuclear power unit needs to be part of the electricity generation installation structure. This is required for the energy security but also to meet the goal of minimizing greenhouse gas emissions.

The strategic plan includes the activities for a second lifetime extension of the existing nuclear power unit in operation – for an additional 10 years from 2026 to 2036. Before 2036 a new nuclear power unit as replacement capacity needs to be commissioned.

The new nuclear unit should be a PWR or an SMR unit based on the pressurized water reactors' technologies. The country's nuclear energy infrastructure is best placed to accommodate a new nuclear unit (or units) based on the pressurized water reactor technology. Drastic change of technologies, compared to operated WWER technology, is not acceptable or, at least, not desirable.

A mid-sized power nuclear plant of about 600 MWe is under consideration to be deployed. Currently 3 possible technological options could be realistic for Armenia: 1) A mid-size PWR unit of about 600 MWe capacity; 2) 2 small size PWR units of about 300 MWe capacity each; 3) Several SMRs. It is a fact that small modular reactors (SMRs) often require adjustments or new developments for the fuel cycle. Many studies have shown that SMRs are incompatible with existing nuclear waste disposal technologies and concepts. Advanced fuels proposed for SMRs may pose different challenges, including the long-term management of used fuel and radioactive waste.

2. DISCUSSION

According to the results of several studies on the implications of SMRs for the back end of the nuclear fuel cycle and on the radioactive waste stream characterization, the SMRs will produce more voluminous and chemically/physically reactive waste than PWRs, which will impact the options for the management and disposal of this waste. It is anticipated that the volumes of spent nuclear fuel also will significantly increase in case of SMR implementation.

As Armenia consider implementation of SMRs, as one of the options of future electricity generation installations, the concepts of fuel cycles for SMRs are important since it can create challenges in spent fuel and radioactive waste management.

The spent fuel from the existing nuclear power unit in operation, after unloading from reactor core, is stored for several years in spent fuel pools, then transferred to dry storage facility for long-term storage.

The SNF dry storage facility design is based on the NUHOMS standard design – version NUHOMS-56V. The NUHOMS system provides storage of SNF assemblies in dry, horizontally placed canisters. The canisters, in turn, are located in storage modules made of reinforced concrete. The modules are arranged in two rows, each of them is closed by an armoured door (see Figure 1).

Removal of residual heat during storage is carried out by natural air circulation and heat exchange through the walls and roof of the horizontal storage module (HSM). HSM is a small reinforced concrete structure designed for normal and extreme loads.

The first stage of the SNF dry storage facility was provided for the storage of 616 pcs. of spent fuel assemblies that remained at the ANPP after the collapse of the USSR and were not taken out for processing. In connection with the collapse of the USSR the design scheme for the shipment of spent fuel was violated.

The filling of the 11 HSMs of the first stage was completed in the period from August 2000 to April 2004. To accommodate the spent fuel assemblies unloaded from the reactor core after the resumption of operation, it was decided to expand the existing storage facility and build the second stage of the storage facility.

The first part of the second stage was put into operation in 2008. It is currently full. The second part of the second stage was put into operation in 2016. Currently, out of 12 HSMs, 9 are filled.

Thus, as of December 2021, 32 out of the 35 HSMs available at the dry storage facility were filled. The number of spent fuel assemblies stored at the facility is 1792.

FIG. 1. External view of ANPP SNF dry storage facility.

In 2023, it is planned to build the third stage of the dry storage facility, consisting of 12 modules. To accommodate all spent fuel assemblies, considering the plant operation till 2036, it will be necessary to build another fourth stage of the spent fuel dry storage facility before 2035. The project of the fourth stage is under discussion. Apparently, dual-purpose vertical containers will be used there.

Management of spent nuclear fuel is one of the most important issues to be solved during the planning of deployment of new nuclear technologies in Armenia. In the process of consideration of the different options of new nuclear technologies, another key aspect is the characteristics of the fuel cycle from the point of view of anticipated radioactive waste streams. The requirements and constraints for spent fuel and radioactive waste intermediate and final disposal applicable to SMR fuel cycle are within the critical factors when making decision on deployment of new nuclear technologies in the country.

Another key aspect is the levelized cost of electricity (LCOE) which, in turn, is directly or indirectly influenced by the fuel cycle concept. When specific designs of SMRs are considered for deployment, detailed study of higher mentioned aspects will be of high priority.

Paper ID#18

DIRECT RECYCLING OF SMR SPENT FUEL FOR URANIUM UTILIZATION IMPROVEMENT AND REDUCTION OF HIGH-LEVEL NUCLEAR WASTE

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Abstract

Small Modular Reactors (SMRs) promise flexible power generation for a wide range of users and applications with high safety features and proliferation resistance. LWR type SMRs are potentially deployable in the near term. Because of its small core volume, SMR cores have higher neutron leakage than that of the large-scale reactors. As a case study recycling of the spent fuel from a NuScale SMR in a CANDU-6 reactor was investigated. Calculations of the depletion histories of the standard and NPM poisons-free and poisoned fuel assemblies were made and shown.

1. INTRODUCTION

Due to their flexible designs, Small Modular Reactors (SMRs) promise flexible power generation for a wide range of users and applications with high safety features and proliferation resistance. SMRs designs rang from scaled down versions of existing reactors to generation IV reactors. SMRs based on LWRs are potentially deployable in the near term. Because of its small core volume, SMR cores have higher neutron leakage than that of the large-scale reactors. It follows that, SMRs will discharge the spent fuel with higher residual fissile material which would: (1) degrade the uranium utilization, (2) increase high-level nuclear waste and (3) enhance the quality of plutonium which decrease the proliferation resistance.

The extraction of fissile isotopes from the spent fuel through the chemical reprocessing and recycling them in the reactors was proposed early when the commercial power plants were built. Over the past three or four decades, some countries like United States of America stopped reprocessing as a political consideration related to weapons proliferation. Also, chemical reprocessing does not have an important impact on reducing the high-level waste. The alternative recycling process is direct use of the spent fuel through dry processing technique such as the Direct Use of Spent PWR Fuel in CANDU Reactors (DUPIC) which is originally proposed in Korea [1]. The general method of DUPIC is re-fabricating the PWR spent fuel in a DUPIC plant to make CANDU bundles. An addition fuel burnup of around 12 GWd/t can be obtained from recycling typical Westinghouse PWR spent fuel in a CANDU-6 reactor. The complex DUPIC processing facility and the transportation of highly radioactive materials from the PWR to this facility and from this facility to the CANDU reactor makes estimation of the economics of DUPIC cycle difficult. However, recycling spent fuel from the PWR-SMR makes the DUPIC cycle more attractive because of the high residual fissile material in the spent fuel.

2. DISCUSSION

As a case study recycling of the spent fuel of NuScale SMR in CANDU-6 reactor was investigated. The NuScale Power Module™ (NPM) is a small pressurized-water reactor with a thermal power of 200 MWth and an electrical power of 60 MWe. The core configuration of the NPM consists of 37 fuel assemblies. The fuel assembly design is modelled on a standard 17 x 17 PWR fuel assembly with 24 guide tube locations for control rod fingers and a central instrument tube. The assembly length is 200 cm, and the fuel is UO_2. The equilibrium loading consists of assemblies with 4.05 % and 4.55 % enriched U-235, the latter of which contains Gd_2O_3. The parameters were defined to be 8 % Gd in 16 rods in the fuel assembly. The core cycle is 24 calendar month and discharge the spent fuel with average burnup of around 40 GWd/t [2]. NPM is in the stage of regulatory review.

A standard Westinghouse 4-loop PWR was consider as a reference case. The reactor core is loaded by 193 fuel assembly with active length of 366 cm. This standard reactor is reflected by light water while the NPM is reflected by stainless steel. MCNPX computational code [3] based on the ENDF/B-VII was used in the calculations. All data of NPM needed for the simulation was captured from Ref. [2].

In the burnup calculations, the lattice cell of the standard and NPM poisons-free and poisoned fuel assemblies were modelled. A criticality calculation (KCODE calculations) with the BURN cards is used to calculate the system criticality and the burnup of the fuel and fuel inventory after each time interval (defined in the BURN cards). The calculations of the depletion histories of the standard and NPM poisons-free and poisoned fuel assemblies are shown in Figure 1.

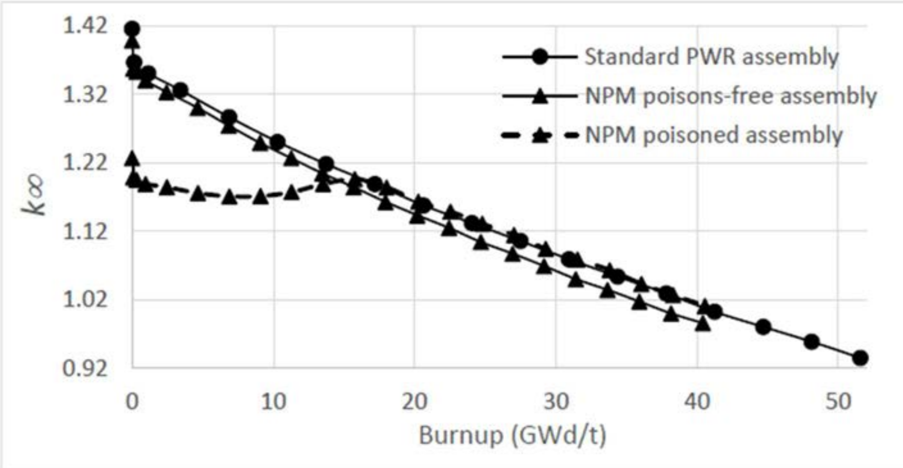

FIG. 1. Depletion histories of the standard and the NPM fuels.

The standard and NPM reactors employ three batch core fuel strategy. At the fuel discharge, the infinite multiplication factors are 0.9348, 0.98566, and 1.01049 for the standard and the NPM poisons-free and poisoned fuel lattice cells, respectively as shown in Figure 1. The differences between the standard fuel assembly reactivity and that of NPM at the discharge is due to the higher residual amounts of the fissile materials in the NPM which would compensate the higher leakage of the NPM core.

TABLE 1. MAIN FISSLE NUCLIDES CONCENTRATIONS IN THE FUEL AT THE LOADING AND AT THE DISCHARGE

Nuclide	**At loading**			**At discharge**		
	Standard (a/b-cm)	**NPM poisons-free (a/b-cm)**	**NPM poisoned (a/b-cm)**	**Standard (a/b-cm)**	**NPM poisons-free (a/b-cm)**	**NPM poisoned (a/b-cm)**
U-235	0.001048	0.000943	0.001055	0.0001789	0.0002274	0.0003025
Pu-239	-	-	-	0.0001279	0.0001243	0.0001311
Pu-241	-	-	-	0.0000376	0.0000314	0.0000318
Total	0.001048	0.000943	0.001055	0.0003444	0.0003831	0.0004654

In the recycling of NPM spent fuel in CANDU-6 reactor, it was assumed that three-fourth of the fuel is that originally poisons-free (4.05 % U-235) and one-fourth was originally poisoned with Gd2O3 with enrichment of 4.55 % U-235. Also, it is assumed that the standard PWR fuel is recycled in the standard CANDU-6 fuel bundle while NPM fuel is recycled in the CANFLEX fuel bundle that order to achieve a higher fuel burnup. After one year of cooling, the material inventories of the standard PWR and NPM spent fuel were input in the CANDU-6 standard and CAFLEX fuel bundles, respectively for the lattice cells calculations. The depletion histories of the recycled standard PWR and NPM spent fuel as well as the natural UO2 fuel, in a CANDU-6 reactor are shown in Figure 2.

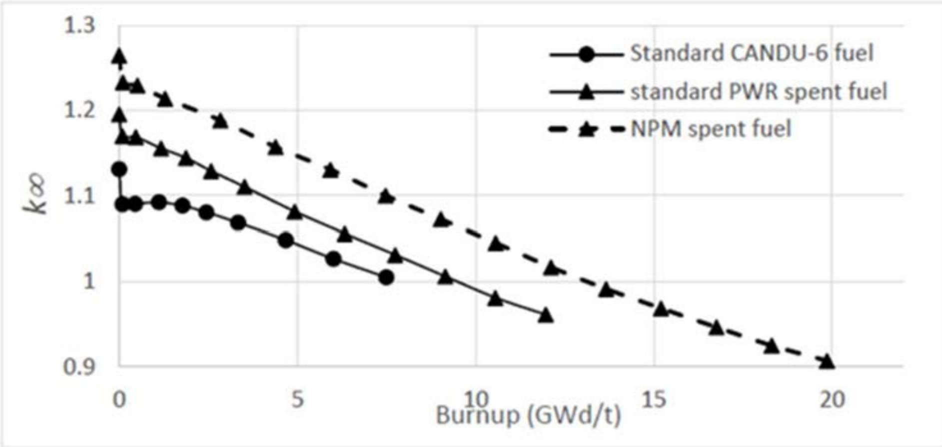

FIG. 2. Depletion histories of the standard CANDU-6 fuel and the standard PWR and NPM spent fuels.

The equilibrium CANDU core contains fuel from fresh to discharge burnup since it is fuelled on-power. CANDU-6 reactors discharge fuel with a burnup of 7.5 GWd/t. The procedure given in Ref. [4] was followed to calculate the fuel burnup of the standard PWR and NPM spent fuel and it is found to be 12 and 20 GWd/t, respectively as given in Table 2. The standard PWR is loaded with 4.5% U-235 and discharges the spent fuel with burnup of 51.6 GWd/t and if it is recycled in CANDU-6, the spent fuel will be burntby an additional 12 GWd/t and the total burnup will be 63.6 GWd/t, an increase of around 23%. In the case of NPM, the fuel has average enrichment of 4.175 % U-235 and is discharged it with a burnup of 40.4 GWd/t. If the NPM spent fuel is recycled in CANDU-6, it will give an additional burnup of 20 GWd/t and the total burnup will be 60.4 GWd/t, an increase of around 50%. Also, the high-level nuclear waste will be reduced by the same percentage i.e., 23% and 50% reduction of the nuclear waste of the standard PWR and NPM reactors, respectively.

TABLE 2. SUMMARY OF CALCULATION RESULT

Reactor type	**Average Enrichment (% U-235)**	**Total burnup (GWd/t)**		**% increase of burnup and reduction in the nuclear waste**
		Without recycling	**With recycling**	
Standard PWR	4.5	51.6	63.6	23
NPM	4.175	40.4	60.4	50

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Paper ID#19

DESIGN EXTENSION CONDITIONS FOR SPENT FUEL STORAGE AT THE PWR NPP KRŠKO

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Abstract

The decision of Slovenia to improve the safety in the Krško Nuclear Power Plant resulted in the upgrade of the Storing of the spent fuel. The original solution for storing spent fuel in the Krško NPP is a spent fuel pool, and the proposed safety modifications were a Spent Fuel Pool Alternative Cooling, completed in 2020, and a spent fuel dry storage (SFDS), expected to be finished in 2022. The transfer of spent nuclear fuel from the SFP into the SFDS is expected to start in 2023.

1. INTRODUCTION

Following the Fukushima accident Slovenia decided to improve safety with regard to severe accidents of its only nuclear power plant, the Krško NPP. The Krško NPP prepared the Safety Upgrade Programme (SUP) for modernization including plant upgrades [1] to address Design Extension Conditions (DEC). Spent fuel in the spent fuel pools (SFP) was affected in the Fukushima accident. Storing of the spent fuel in the Krško NPP also had to be upgraded and was considered in the SUP. The original solution for storing spent fuel in the Krško NPP is spent fuel pool. Its capacity was increased to 1694 spent fuel assemblies in 2003. However, the increased capacity is not big enough to ensure storing of spent fuel until the end of the Krško NPP long term operation in 2043. The Krško NPP analyzed the situation and found out that the optimal solution is to build the spent fuel dry storage (SFDS). Therefore, safety upgrade of storing spent fuel pool included the existing SFP and building of the SFDS.

2. DISCUSSION

The SFP is located within the Fuel Handling Building at the Krško NPP site. It is a reinforced concrete structure and designed according to the seismic and other criteria for safety structures. The Spent Fuel Pool Alternative Cooling is a safety modification for the SFP. The modification includes installation of a fixed spray system on the spent fuel pool with provisions to use mobile equipment and acquisition of a mobile heat exchanger, which is located outside the nuclear island and with provisions for quick connections to the SFP. The modification was completed in 2020. If design basis heat removal system fails, the SFP can be cooled under DEC A conditions, using mobile heat exchanger. It can remove 8.5 MW of heat load and keep the SFP water temperature below 80ºC. The mobile heat exchanger can be cooled in two ways, using water from the fire protection system or the Sava River with provisions for quick connection to the SFP.

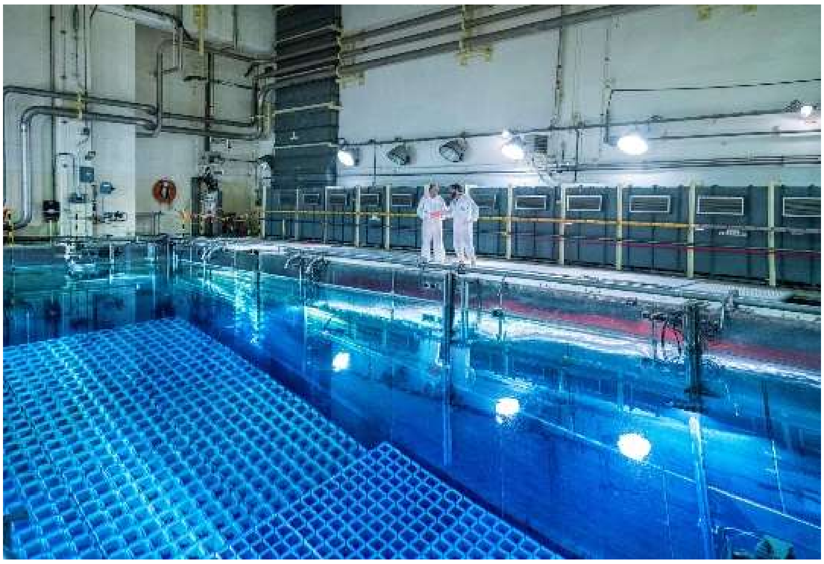

FIG. 1. Krško NPP Spent Fuel Pool (reproduced from Ref. [1] with permission).

The SFDS is being constructed within the NPP Krško site. It will have a capacity of up to 2,590 spent fuel assemblies in 70 casks, type HI-STORM FW from HOLTEC. The decay heat removal in the SFDS is passive. Ground floor dimensions of the SFDS are 69.80 m by 47.70 m. The maximum height above the terrain will be 20.48 m. The SFDS building is a reinforced concrete and steel structure with a primary function to store the HI-STORM FW system on the pad. The concrete structure of the building with additional steel plates installed inside the building provides additional shielding so that the building and HI-STORM FW system meet the site boundary dose requirements for independent storage of spent nuclear fuel at the Krško NPP site. The SFDS includes a handling area, a technical area, a storage area and a cask transfer facility. The SFDS is designed for Design Extension Conditions (DEC) applicable to the earthquake with peak ground acceleration (PGA) = 0.78 g. It can also withstand extreme atmospheric conditions (temperature, humidity, glaze ice), strong wind (up to 240 km/h), flooding, fire, and airplane crash. The design criteria are presented in more detail elsewhere [2, 3]. The construction of the SFDS is expected to be finished in 2022. The transfer of spent nuclear fuel from the SFP into the SFDS is expected to start in 2023.

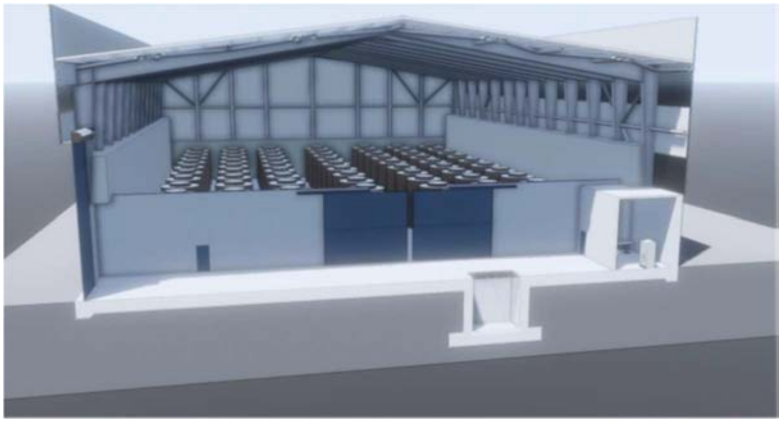

FIG. 2. 70 casks can be stored in the SFDS (Krško NPP archive) (reproduced from Ref. [5] with permission).

The SFDS is designed, and casks are chosen considering the properties of spent fuel from the Krško NPP. Slovenian legislation requires [4] that acceptance criteria for the storage or disposal of spent fuel has to include the limits for a number of parameters, namely: the content of radionuclides and specific activity, dose rates on the surface and at reference distances from the package surface, specific surface contamination, strength, leachability, corrosiveness, chemical stability, heat generation, degradation effects of radiation, i.e. changes in the properties of materials resulting from exposure to ionizing radiation, change in volume, flammability, gas formation and gas content, content of toxic substances, content of organic substances with potential effects on microbiological degradation, free liquid content, presence of chelating and other complexes, explosiveness, combustibility, corrosion resistance, permeability and porosity, void fraction, criticality, suitable method of the labelling of radioactive waste or spent fuel packages and adequacy of the packaging and the method of packaging radioactive spent fuel. The Krško SFDS can in principle be used for storing spent fuel from other reactor types. It means that, in principle, it can also be used for storing spent fuel from SMRs. However, before that it should be proven that spent fuel from SMRs fulfils the above acceptance criteria for storing in the Krško SFDS. We should not be surprised if some of the above acceptance criteria are not fulfilled. SMRs can have various enrichment of fuel and operating conditions.

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Paper ID#20

THE PROCESS OF DECARBONIZATION OF THE DOMESTIC POWER INDUSTRY IN POLAND USING SMR REACTORS

Issued related to spent fuel as one of the parameters determining the choice of technology

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Abstract

Polish energy policy aims at achieving climate neutrality by changing the structure of using primary energy sources, gradually moving away from the use of fossil fuels in favour of the exploitation of low-emission sources. The nuclear reactor technology most suitable for replacing coal units is based on high temperature reactors, most of which are classified as SMRs. Bearing in mind the interactions of individual factors related to the management of spent fuel of the reactors, the assessment of the nuclear technology that could be used in the process of the decarbonization of the domestic power industry in Poland could be challenging and some compromises will probably be required.

1. INTRODUCTION

The power sector is a large source of global greenhouse gas emissions and coal-fired power plants are the most significant source of emissions in this sector. Rapid decarbonization of coal-based economies, including Polish industry, is essential to avoid the worst effects of climate change. Polish energy policy aims at achieving climate neutrality by changing the structure of using primary energy sources. There is consensus that Poland needs to gradually move away from the use of fossil fuels in favour of the exploitation of low-emission sources. When selecting a decarbonization path, it should be considered whether a given method is technically feasible and economically viable. A new and very promising direction of decarbonization of power industry is its modernization towards the use of nuclear reactors [1-3]. The nuclear reactor technology most suitable for replacing coal units is based on high temperature reactors, most of which are classified as SMRs (small modular reactors). In addition to Generation III reactors, the group of Generation IV reactors is also considered as a potential nuclear technology of future. When choosing a nuclear technology, technical issues are very important, e.g., reactor power, the temperature of generated steam and fuel properties [4]. However, other considerations are also important and should be taken into account. An important factor that should be considered is the amount of spent fuel produced and the method of its management. The issues of managing SMR spent fuel during storage, transportation, possible reprocessing and recycling, or final disposal are obviously connected with the nuclear safety of the whole process and also with its economy.

2. DISCUSSION

The multitude of SMR technologies is clearly related to the use of various types of fuel [4] as it is shown in Table 1. The amount of spent fuel, its chemical and radiochemical form, total activity, as well as the type of radioactive waste generated during the operation of different types of reactors can vary greatly. The method of storage, transport and final disposal of these materials depends on all these factors.

TABLE 1. SMR DESIGNS WHICH ARE CONSIDERED AS POSSIBLE RETROFIT OF COAL-FIRED PLANTS FOR THE POLISH ENERGY SYSTEM UNDER DESIRE PROJECT [4]

Design	**Fuel type**	**Fuel enrichment %**	**Coolant**	**Power**	**Temperature**
High-Temperature Gas-cooled Reactor (HTR-PM)	TRISO fuel	<19,75%	Helium	250 MWth	750°C Helium 566°C steam
Xe-100 — X-Energy (HTGR)	TRISO X UCO, UO_2	5-20%	Helium	200 MWth	750°C Helium 565°C steam
ThorCon molten salt fission reactor	Uranium and 9.0 kg of thorium per day	19.75%	salts	557 MWth	550 °C
Kairos Power FHR (KP-FHR)	Uranium oxide (UO_2)	19,75%	salts	320 MWth	585 °C
Terrestrial Energy Integral Molten Salt Reactor (iMSR)	Molten UF_4	<19,75%	salts	400 MWth	~ 585 °C
Westinghouse Lead Fast Reactor (LFR)	Oxide (UO_2 or MOX prototype); Advanced, high-density fuel (commercial)	Fast reactor	Lead	950 MWth	~ 595 °C
NuScale (SMR-PWR)	Uranium oxide (UO_2)	8-9%	Water	250 MWt/unit	288°C

In the approach, implemented in the design project, a ranking of nuclear reactor technologies is created. Issues related to management of the SF as well as RW, in addition to those associated with the safety systems of the reactor itself, the heat cycle of the steam turbine and additional infrastructure, are important matters affecting the assessment of the competitiveness of a given technology [5-7]. Several parameters concerning SF and RW management will be assessed, the most important are shown in Figure 1.

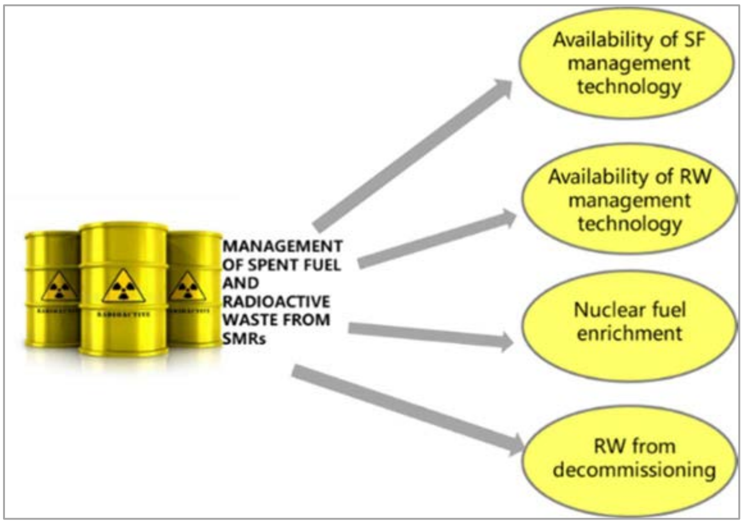

FIG. 1. Issues related to SF and RW management that need to be taken into consideration when selecting the SMR technology.

Reactor technologies for which methods of dealing with SF and RW have already been developed, will be rated higher than the others. Technologies with known methods of processing, transport and final storage would be rated the highest, while those for which none of these management stages has yet been developed would score the lowest. Another parameter to be assessed is the degree of nuclear fuel enrichment, which translates, among others, into the amount and radioactivity of the generated SF and RW. Several designs of Gen IV reactors are considering the use of high-assay low-enriched uranium (HALEU) fuel; however, its applications are limited to the production of small batches for research reactors and medical radioisotope production. Additionally, the supply of HALEU fuel requires improvements in the current nuclear fuel cycle infrastructure to comply with potential criticality safety limits, in particular the development of enrichment, de-conversion, and fabrication facilities [4]. While LWR-SMRs are expected to use the low-enriched fuel, it results in lower thermal efficiency and directly impacts the fuel cycle costs. Another parameter under evaluation is the amount and type of waste generated during the decommissioning phase of the nuclear facility. Technologies that generate during the decommissioning only typical construction waste and "light" water, will be rated the highest, while those, which generate problematic waste, such as irradiated graphite, molten salts, etc. will be ranked the lowest. The problematic/characteristic wastes are currently the subject of much research in order to optimize methods for their safe storage, treatment and final disposal [8,9]. Moreover, the quantities of characteristic wastes may be reduced by continuous improvements in optimum use of nuclear reactor materials during the design stage.

Bearing in mind the interactions of individual factors related to the management of SF discussed above, the assessment of the nuclear technology that could be used in the process of the decarbonization of the domestic power industry in Poland could be challenging and some compromises will probably be required.

ACKNOWLEDGEMENTS

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Paper ID#21

CONSIDERATIONS AND PERCEPTIONS FOR NUCLEAR FUEL CYCLE BACK END RELATED TO THE SMRS UNDER CONSIDERATION IN JORDAN

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Abstract

Jordan has limited access to natural resources, especially water and energy, and thus far has been facing significant challenges in planning a stable energy future. The Jordan Nuclear Strategy pursues two parallel tracks: Small Modular Reactor and Large Nuclear Reactor. Jordan has no final decision on the SNF treatment as it will be based on the selected technology. SMRs under Consideration in Jordan are summarized.

1. INTRODUCTION

Jordan became a member of the International Atomic Energy Agency (IAEA) in 1966 and has started close cooperation with the IAEA to prepare for a safe and secure nuclear programme. In November 2006, a high-level ministerial committee chaired by the Prime Minister was established to develop a roadmap for implementing the nuclear energy programme, this committee set up the Nuclear Energy Programme Implementing Organization (NEPIO).

In 2007, Law No. 42 was issued to establish the Jordan Atomic Energy Commission (JAEC) and empowering it to "lead the development and implementation of nuclear strategy and to manage the nuclear energy programme". One of the primary responsibilities, is "the development and eventual deployment of commercially viable nuclear power plants for energy generation in Jordan. This deployment comes as part of Jordan's efforts for energy diversification, and the implementations of Jordan's concurrent energy strategies where nuclear power generation will be an integral component" [1].

Jordan has limited access to natural resources, especially water and energy, and thus far has been facing significant challenges in planning a stable energy future. Available energy options in Jordan are limited: Natural Gas, Renewable Technologies, Oil Shale, and Nuclear Energy.

2. DISCUSSION

The Jordan Nuclear Strategy pursues two parallel tracks: Small Modular Reactor and Large Nuclear Reactor. JAEC has been considering SMRs and their potential in Jordan in various capacities since 2011. The advantages of SMRs are many for Jordan, but it has always been the maturity of the technologies for deployment, and constructability at a competitive price that were the hindering factors. In deciding to proceed forth today, JAEC took into consideration many factors that are bespoke to SMRs [2].

The National Policy for Radioactive Waste & Spent Nuclear Fuel Management serves as a national commitment to address the management of spent nuclear fuel (SNF) and radioactive waste (RW) in a coordinated and cooperative manner with all related national organizations and entities, for Jordan poised to enter the era of nuclear energy for the nuclear peaceful applications to be in accordance to the national energy strategy off 2006 The aim of the national policy for the management of spent nuclear fuel and radioactive waste generated from the nuclear fuel cycle is to manage it in a safe manner with full commitment to not cause any hazard or damage to the environment in the short term or in the long term The policy includes all specific steps and stages related to the safe management of the spent nuclear fuel and radioactive waste; "starting from its generation phase to its final disposal phase, to ensure the nuclear safety and security provisions that aims to protect human health and the environment against the radiological and nuclear contamination, currently and in the future and without imposing undue burden upon the next generations" [3].

The National Policy for Radioactive Waste & Spent Nuclear Fuel Management is to store the spent nuclear fuel on an interim basis in pool storage at the nuclear power plant site until it decayed to sufficient levels to allow for safe storage, establishing storage facilities near the nuclear power plant for further cooling.

Consideration is being given too decide on the possibility of returning the spent nuclear fuel to the country of its origin to the supplier for final disposal or interim storage or to remain in Jordan in interim storage for one of the two following options [1]:

— Considering the spent nuclear fuel as strategic resource that can be utilized through reprocessing (nationally or internationally) where the subsequent HLW will be sent for final disposal in a national waste disposal facility at The Hashemite Kingdom of Jordan;
— Declare that the spent nuclear fuel as radioactive waste thereby allowing it to be disposed of directly to a national waste disposal facility.

Work is underway to establish the national facilities for disposal of low and intermediate level waste (LILW) in the Hashemite Kingdom of Jordan and to take relevant decisions for the disposal of spent nuclear fuel and high-level waste (HLW) [3].

The National Strategy for Radioactive Waste & Spent Nuclear Fuel Management serves as the basis for the facilities that generate and manage RWs to draft their own institutional strategies, and to achieve a harmonized level of RWM in Jordan. It also presents the actions needed to implement Jordan's radioactive waste policy that was published in 2015. The strategy presents technical management options along with the needed national infrastructure for present and future RWM activities.

The National Strategy for Spent Nuclear Fuel Management defines the safe interim storage option:

— Spent fuel discharged from the reactor will be safely stored in spent fuel storage pools until the decay heat generated from the spent fuel no longer requires water cooling;
— Sufficiently decayed spent fuel will be relocated to dry storage facilities built on the reactor site.

After storage at the reactor, different technical options will be considered, as follows:

— Return of the spent fuel to the country of origin for processing and final disposal (the take-back approach). This option is called "Return to Manufacturer";
— Shipment of the spent fuel outside Jordan for reprocessing, and the HLW generated from reprocessing will be shipped back to Jordan for final disposal in a deep disposal facility. This option is called "Outside Jordan";
— Spent fuel will be stored in Jordan to allow for its direct disposal at the national deep geological disposal facility in Jordan.

Both latter options require JAEC to construct and commission a Deep Geological Disposal facility and conditioning facility with associated infrastructure (facility to condition HLW and/or SNF before final disposal in Deep Geological Disposal) [4].

The primary objective of the BIS is to "provide information to the bidders (the prospective contractors) on number of topics. It is in the interest of the owner to provide complete and precise information in the BIS since this will facilitate the preparation and subsequent evaluation of the bids. It is also in the interest of the owner to promote competition and to encourage each bidder to prepare and present a bid that meets the owner's needs in the best possible way. This means that the owner should provide comprehensive information on all aspects which may affect the project and the owner should clearly express the requirements, conditions and wishes or expectations. On the other hand, the owner should refrain from being too restrictive in the demands and from making the content of the BIS too extensive by including detailed technical descriptions or basic information which is common knowledge" [5].

Technology Assessment in Jordan of selected SMR technologies is being conducted in two main phases: "The first phase will be the generic assessment phase with the aim of down-selecting the most advanced and competitive technologies that are deployable and viable in Jordan. The next phase will be the preparation of a Feasibility Studies (FS) based on the short-listed technologies or issuance of BIS. As per the results of the Assessment or FSs, a Justification of Investment analysis will be made to proceed forward with the selected SMR."

The differences between technologies and their impact on Jordan will be assessed through rigorous evaluation methodologies designed to bring full visibility and transparency:

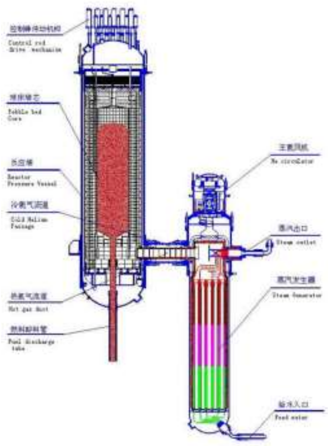
- Assessment of the vendor technology towards Key Factors (important for Jordan);
- Evaluation Matrix;
- Best-in-Class for each evaluation criteria;
- Price under competitive environment [2].

SMRs under Consideration in Jordan can be summarized as shown in the following Figure 1.

ACP-100S Chinese iPWR

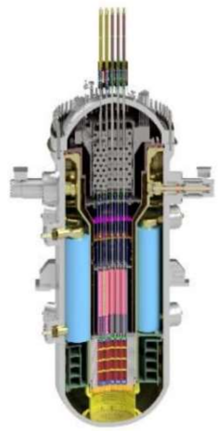
- 125 MWe (Gross) / module
- 112.5 MWe (net) / module
- 0.3 g seismicity
- 2 passive safety trains
- 24 month refuelling cycle

HTR-PM Chinese HTR



- 110.5 MWe (Gross) / module
- 103 MWe (net) / module
- 0.3 g seismicity
- 2 passive safety trains
- Online refuelling

SMART South Korean iPWR



- 123 MWe (Gross) / module
- 103 MWe (net) / module
- 0.3 g seismicity
- 4 passive safety trains
- 30-36 month refuelling cycle

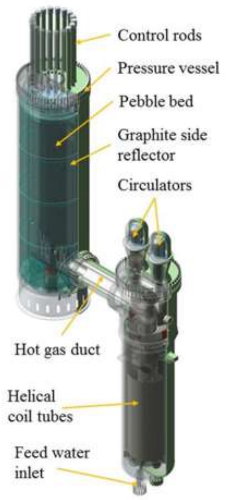
RITM-200 Russian iPWR

- 57 MWe (Gross) / module
- 52.5 MWe (net) / module
- 0.3 g seismicity
- 4 safety trains
- 48-72 month refuelling cycle

NuScale American iPWR

- 77 MWe (Gross) / module
- 74 MWe (net) / module
- 0.5 g seismicity
- 2 passive safety trains
- 24 month refuelling cycle

Xe-100 American HTR



- 81.5 MWe (Gross) / module
- 75 MWe (net) / module
- 0.5 g seismicity
- 2 passive safety trains
- Online refuelling

FIG. 1. SMRs under Consideration in Jordan.

Three SMRs (based on matrix evaluation criteria) have been shortlisted:

— HTR-PM: The spent fuel elements are transferred to the designed storage tank, the capacity of which is 40 000 pebbles, the filled storage tanks will be placed in shielded concrete compartments in the spent fuel intermediate storage [6];

— NuScale: After removal from the reactor core, used fuel assemblies are placed in dedicated Used fuel storage racks in the below ground Used fuel pool After cooling in the spent fuel pool, spent fuel is placed into certified casks, steel containers with concrete shells, on site of the plant. The facility is designed for ease of Used fuel transfer to a dry cask storage system [7];

— RITM: Firstly, the spent fuel follows to a wet storage, where leak-tight tanks are used, and where decay heat removal is performed. Afterwards, fuel will be transferred to a dry storage, where leak-tight canisters are used [8].

Jordan has no final decision on the SNF treatment as it will be based on the selected technology.

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Paper ID#22

BACK END OF THE FUEL CYCLE CONSIDERATIONS FOR SMALL MODULAR REACTORS

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Abstract

With the objective of achieving carbon neutrality by 2050, the installation of SMRs and their technology should be part of the French energy strategy. In that matter, the French SMR – Nuward and new reactor concepts in the field of fission and fusion are promoted.

1. INTRODUCTION

To reach carbon neutrality by 2050 in order to limit global warming, an energy transition has been initiated in France. Renewable and nuclear energies can play a significant role in a context of electrification of many uses. Nuclear energy development requires construction sites, financial commitments of several billion euros and societal approval. Nuclear projects are therefore complex to bring about. Small nuclear reactors, which require lower financial commitments, with shorter construction times and with greater simplicity to operate should be part of the French energy strategy [1].

2. DISCUSSION

In this context, the investment plan « France 2030 » will dedicate 500 M€ to the development of the French SMR – Nuward and additionally 500 M€ for proposals of new reactor concepts in the field of fission and fusion [2].

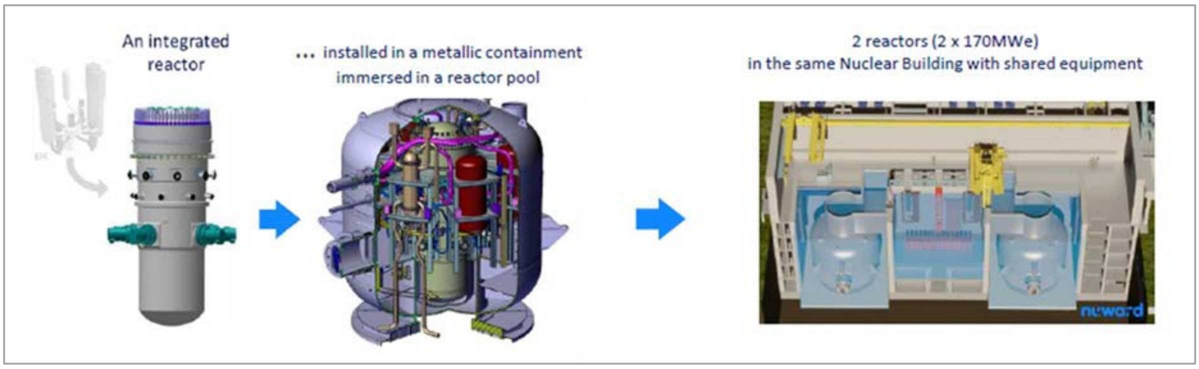

FIG. 1. NUWARD (reproduced from Ref. [3] with permission).

The main objective is to create a new ecosystem for the nuclear sector. The collaboration of the French nuclear authority with the Finnish and Czech safety authorities for a joint preliminary review of the NUWARD™ reactor project is a good example of this new ecosystem [4]. France has all the facilities needed for the spent fuel recycling even if some modifications could be needed to accommodate new fuels coming from the SMR. However, before taking the decision to launch the development of new nuclear reactor technologies, a global vision of the cycle is needed: starting materials and natural resources, life cycle, economy, nuclear transportations, wastes management…). The question of the market aimed by the provider of the national SMR project (i.e., domestic or foreign) is raised.

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Paper ID#23

ADVANCED NUCLEAR REACTORS: WHAT ABOUT THE BACK-END?

Focus on reprocessing/recycling on treatment of various fuel types

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Extended Abstract

Since the 1950's, the nuclear power industry has been continuously developing and improving reactor technology seeking for even more safety, reliability, and economic competitiveness. Today, nuclear power is experiencing a renewed interest around the world and a consensus emerges on the critical role that it needs to play in the fight against climate change by offering a huge source of low-carbon, competitive, constant, and controllable energy for electricity, heat, and potentially hydrogen production.

Besides conventional nuclear units that will continue to play a major role in power supply, many concepts of Advanced Reactors are being developed to provide new solutions for the future. There is particularly strong interest for small or medium reactors (SMR/AMR), which can propose simpler units for generating electricity from nuclear power, and for process heat. The proposed technologies are numerous and very diverse [1].

However, nuclear systems will be able to offer fully sustainable solutions only if they take care of the back end of their fuel cycle. Ensuring that used/spent nuclear fuel (UNF/SNF) produced by these Advanced Reactors is safely, securely, and economically handled and prepared for post-reactor activities (e.g., fuel conditioning, interim storage, and transportation for downstream management) is critical and may even be compulsory to get license for these new reactors. From a sustainability point of view, closed fuel cycle, when it is possible, offers many advantages in terms of waste minimization, preservation of resources, and reuse of valuable materials, but alternatives exist for SNF that cannot be easily recycled [2].

All these concepts of Advanced Reactors can be approached and categorized by the type of fuel they use: oxide/ceramic fuel (with special case of TRISO fuels), metal or alloy fuels, and liquid fuel salts, with different fissile materials and multiple levels of enrichment (for U fuels). The back end of the fuel cycle for these reactors needs to be considered as early as possible in the design to account for the potential economic, technological, and regulatory challenges that they present.

Oxide/ceramic fuel with metallic cladding benefit from the extensive experience gained in large scale power reactors, both in terms of manufacturing, irradiation, transportation, reprocessing and recycling (to date, more than 39,000 tons of used fuels have been reprocessed at La Hague plant and more than 3,000 tons of MOX fuel have been fabricated at Melox plant [3]). Regarding future reactors, specific features are considered to make better use of their fuel (like higher initial enrichment) or include technological and safety advances (like Accident-Tolerant Fuels). Even if some adaptations are needed, no showstopper is foreseen to offer complete back-end solution for these advanced fuels.

TRISO fuels, either embedded in pebbles or in cylindrical compacts, have been designed to be as inert as possible and to support high temperatures and high pressures to offer a good containment of fission gases. In addition, the quantity of valuable materials that could be recovered from used fuel being very limited, direct disposal seems to be the easiest option for such type of spent fuel. However, specific considerations may lead to reconsider this statement:

— The large volume of spent fuels to be managed in a context where geological repositories will be expensive and scarce resources that need to be used cautiously;
— The acceptability of the irradiated graphite in geological repositories is not ensured especially if such waste forms have not been accounted from the beginning in the design safety case of the facility.

Due to these reasons, a treatment of HTGR spent fuel could be contemplated to separate the bulk graphite (more than 90 %wt. of the fuel) from the TRISO particles. Several technologies have already been explored in the past (thermal, mechanical, or chemical treatment) and the most promising of them could probably be developed up to industrial scale, but a significant work remained to be done to achieve the main requirements associated to such treatment [4]:

— No TRISO particles remaining with the graphite (to ease the management of graphite waste);
— Maintain integrity of TRISO particles during separation: The SiC layer needs to not be broken to prevent release of gaseous fission products and prevent contamination of the graphite;
— Deal with C14 release.

Metallic fuels are essentially considered for fast reactor concepts. The potential presence of sodium metal, whether external as a result of using a sodium coolant or internal by the use of a sodium bond in the clad-fuel gap (to keep the temperature low) create specific issues for back-end management. The presence of a sodium-bond in the clad-fuel gap will make the fuel not suitable for certain repositories, requiring treatment of the fuel prior to disposal.

Aside from metallic sodium, uranium metal that is exposed to moisture has the potential to form pyrophoric uranium hydrides. Uranium hydrides have previously led to pyrophoric events during handling of spent storage cans that were opened in an air environment. Special care needs to be taken when handling the metal fuels in order to prevent these excursions. Safety case for emplacement of significant quantities of such waste form in a geological disposal facility will require some additional development as all development has been based on oxides fuel form so far.

One potential solution to these problems would be to reprocess used metallic fuels. Pyroprocess would probably be the most suitable process to treat metallic used fuels. However, pyroprocessing has not yet reached the same level of industrial maturity than hydro-processing that is currently used for oxide fuels.

For liquid-fuelled molten salts reactors, used fuel circulate in the core in liquid form, offering the possibility of long irradiation cycles (no cladding-related issues) to burn actinides and accumulate fission products. Refuelling can be made by periodic drainage of used fuel compensated by addition of the same quantity of fresh liquid fuel.

Direct disposal of SNF, even after conditioning, does not seem to be a reasonable option for such highly corrosive material. On the contrary, such type of fuel could be well adapted to closed cycle scenarios:

— The use of a liquid fuel without a solid fuel assembly avoids the need to dispose of the structures of the assemblies compacted in ILW waste form, hence contributing to the reduction of the final waste volume;
— Salt-type fuels are suitable for processing: hydro-processing (for chloride-type fuels), that has strong industrial records and can be used after dissolution of the salts in aqueous medium and removal or harmful ions for the hydrometallurgical process and

equipment. Pyro-processing is also an option (and necessary for fluoride-type fuels) but has not reached yet the same industrial level than hydro-processing.

The possibility to efficiently burn actinides in MSR system opens the door to reduce drastically the long-term radiotoxicity and lifespan of HL waste (that will only contain fission products).

Advanced Reactors will play a critical role in the future of nuclear power industry to provide adapted solutions for generating electricity, process heat or hydrogen for green energy needs. Various concepts of AR will probably reach industrial deployment and ensuring that UNF/SNF produced by these Advanced Reactors are safely, securely, and economically handled and prepared for post-reactor activities will be key to allow a sustainable deployment of these new nuclear systems. Back end fuel cycle management has to be considered as early as possible in the design process to account for potential economic, technological, and regulatory challenges that it presents.

In addition, in a world confronted ever more with the finiteness of its resources, sustainability and circular economy are becoming key development drivers in our societies, and even requirements for future generations as they are critical to fight against climate change and preserve our environment. From this sustainability point of view, energy systems able to propose the best use of natural resources and the best solutions for waste management should be preferred. Currently implemented recycling fuel cycle already offers such advantages in terms of waste minimization (volume reduced by 5 and radiotoxicity reduced by 10) and standardization, preservation of natural resources (up to 25% of natural resources saved in LWR mono recycling), and reuse of valuable materials (96% of valuable materials recovered by reprocessing). Well proven solutions based on hydrometallurgy and MOX recycling already offer available solutions for managing back-end of part of advanced nuclear systems (oxide-based fuel) and enables new technologies / processes development to provide efficient solutions for various types of fuels in the future, and thus mitigating the financial risks of the geological disposal. In addition, efficient synergies can be developed like the association of MSR actinide converters with treatment-recycling plants to offer of the most suitable and efficient means to drastically reduce the volume, lifespan, and radiotoxicity of a standardized waste form from various used fuels type,

Even when recycling fuel cycle cannot be implemented, viable and credible solutions for back end management have to be developed and implemented up to the disposal of SNF, classified as waste relying on the deployment of geological repositories and associated conditioning facilities. The development programmes carried out in different countries for the past decades clearly demonstrate that the licensing of such facilities requires strong stakeholders support, and overall cost is much higher than initially anticipated. Thus, such facilities constitute scarce resources that needs to be used sparingly. In that case, efficient solutions need to be developed to optimize the volume of spent fuels to be disposed of, separate and manage the elements that could be non-compatible with geological repositories (like graphite in TRISO fuels for example).

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Paper ID#24

MOLTEN SALT REACTORS TECHNOLOGY: OPPORTUNITIES OF MOLTEN SALT FUEL FOR ACTINIDES MANAGEMENT

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Abstract

French spent fuel reprocessing is based on the closed cycle strategy with the idea that fissile actinides produced in LWR can be reused and valorized, which permits enriched uranium savings. In the framework of a common R&D programme around the fuel cycle and transuranic actinide management, CEA and ORANO launched in 2020 preliminary studies around fast MSR technology. The objective was to assess the economic opportunities, the technical feasibility, and the major R&D issues of a small fast MSR for plutonium conversion. Following the Advanced Reactor for Actinide Management in Salt (ARAMIS) project, the Innovative System for Actinides Conversion (ISAC) project (2022-2025), supported by the *France Relance* recovery plan, has the ambition to assess the potential benefits of fast MSR with respect to minor actinide management

1. INTRODUCTION

In a Molten Salt Reactor (MSR), the fuel is liquid and acts both as fissile material and coolant. The fuel is confined in a closed loop whose boundary acts as the first barrier. The operation of such reactors is far from the well-known operation of traditional solid fuel reactors (Light Water Reactors, Sodium Fast Reactors…). The molten fuel circulates inside a closed loop from the critical area, where the energy is produced, to the heat exchangers, which transfer the energy to the Power Conversion System. The core neutronic reactivity depends on salt temperature, salt isotopes (actinides, fission products, base salt elements) and salt flow which transports the delayed neutron precursors from the critical area to the off-flux area. Local fuel temperatures and fuel concentrations are tightly linked in the core. When the salt cools, it becomes denser, leading to an increase in reactivity and therefore in power. Conversely, when the salt heats up, it expands, causing a drop in reactivity. Neutron feedback effects follow the quick density variation of the liquid fuel. The fuel loop has then a naturally safe behaviour with regards to safety provided that the residual power removal is assured.

This kind of system has been studied since the 1950s with the operation of reactor prototypes in the United States (ARE, MSRE [1]) and recently in China (TMSR-LF1 [2]). They were studied in particular with regard to their increased flexibility in terms of power control, due to the quick thermal feedback effects of liquid fuel. This point is regularly displayed as one of the economic advantages of MSR, in the context of a growing intermittent renewable power share in the energy mix. From 2000s, a renewed international interest in terms of research and development has been observed with various national programme around MSR (MCFR [3] (United States of America-Terrapower), MOSART [4] (Russia-Kurchatov), TMSR-LF1 (China-Sinap), MSFR [5] (France-CNRS)). MSR have the potential following benefits: flexible in terms of power variation, flexible in terms of fuel isotopic enrichment, quick and efficient thermal feedback effects, a fuel loop without pressurization, natural convection capabilities, weak structural materials mass, high efficiency of the energy conversion system, high achievable temperature for direct heat production (> 1000°C). MSR development roadmaps need to resolve various key issues like the corrosion resistance of materials submitted to molten salts, the thermal load due to high temperature operation, the salt nuclear depletion and the fission products management [6], the overall reactor and chemical units' safety [5], the handling of salt and the maintenance of components, and the radioprotection strategy.

2. DISCUSSION

French spent fuel reprocessing is based on the closed cycle strategy with the idea that produced fissile actinides in LWR can be reused and valorized, which permits enriched uranium savings. Actinides of interest (mainly plutonium) are reprocessed and incorporated in new fuels. Fission Products (FP) and Minor Actinides (MA = Am, Np, Cm) are separated to be stored in an inert glass matrix inside deep geological disposal [7]. Mixed Oxide (MOX) fuels which contains both uranium and plutonium are currently used in some of French LWR plants. However, actinide capture cross sections are far lower in thermal spectrum than fissile materials which degrades the plutonium quality and increases the minor actinides content of spent fuel (see TABLE 1). Such spent plutonium is barely reusable in LWR due to high content in plutonium fertile isotopes (Pu-240/242). Some R&D studies are currently performed to overcome this problem in LWRs but the ultimate goal is to use it in future fast spectrum systems [8]. Moreover, LWR are not adapted to transmutation of minor actinides unlike Fast spectrum systems, like SFR or Accelerator Driven Systems (ADS) (see. references [1,8,9,10,11,12]). According to [8], ADS present economic drawbacks when compared to SFR. In this context, we can wonder about the opportunities of fast neutron MSR with respect to transuranic actinides management ([13,14,15]).

TABLE 1. MINOR ACTINIDES INVENTORIES IN SPENT FUEL (AFTER A 5-YEAR COOLING TIME) – RESULTS ISSUED FROM [1]

Isotopes	UOX-LWR 46 GWd/t (g/TWhe)	MOX-LWR 48 GWd/t (g/TWhe)	MOX-SFR 99 GWd/t (g/TWhe - equilibrium)
Np-237	1700	390	460
Total Np	1700	390	460
Am-241	1160	8900	2900
Am-243	540	5100	680
Total Am	1700	14000	3600
Cm-244	190	2400	190
Cm-245	16	420	18
Total Cm	210	2900	215
Total MA	3610	17290	4275

Transmutation is a time-consuming process due to the irradiation time firstly but also due to the cooling time before reprocessing and finally to the reprocessing and fuel fabrication time. Moreover, irradiation time in solid fueled reactors is typically limited by the irradiation damage in clad due to fast neutron irradiation (typical limit around 100 dpa for current materials). With a limited irradiation time, full conversion of whole actinide cannot be achieved within one cycle. A multi-recycling strategy is then mandatory knowing that a typical overall cycle time for a SFR is around 10 to 15 years [10,11]. With molten salt reactors, we have the opportunity to decrease out-of-pile time (simplified fabrication process, pyrochemical process to deal with hot fuel) but also increase in-pile time. Moreover, whereas in SFR MA content is limited by their negative impact on safety thermal feedback effect [8,9], in a molten salt reactor MA content does not affect significantly the major feedback effect caused by the fuel thermal expansion. Fast spectrum MSR offer the possibility to increase MA content and conversion efficiency like in solid fuel ADS systems.

In the framework of a common R&D programme around fuel cycle and transuranic actinide management, CEA and ORANO launched in 2020 preliminary studies around fast MSR technology. The objective was to assess the economic opportunities, the technical feasibility, and the major R&D issues of a small fast MSR for plutonium conversion (Figure 1). Following the Advanced Reactor for Actinide Management in Salt (ARAMIS) project, the Innovative System for Actinides Conversion (ISAC) project (2022-2025), supported by the *France Relance* recovery plan, has the ambition to assess the potential benefits of fast MSR with respect to minor actinide management. ISAC gathers 5 French partners (CEA, CNRS, EDF, FRAMATOME and ORANO), whose purpose is to assess the impact of transmuter MSR on fuel cycle facilities from manufacturing plants to terminal waste disposal facilities ([11,16]). A second goal of ISAC is to consolidate reactor design process and to work on key R&D issues by funding experiments on materials corrosion, salt synthesis, recycling processes and salt base properties.

TABLE 2. MAIN CHARACTERISTICS OF ARAMIS DESIGN

Power	300 MWth
Salt	NaCl-MgCl2-(Am,Pu)Cl3
Core volumic power	250 MW/m^3
Fuel loop number	6
Total fuel salt volume	5,2 m^3
Heat Exchanger type	Shell and Tube
Core volume ratio	25 %

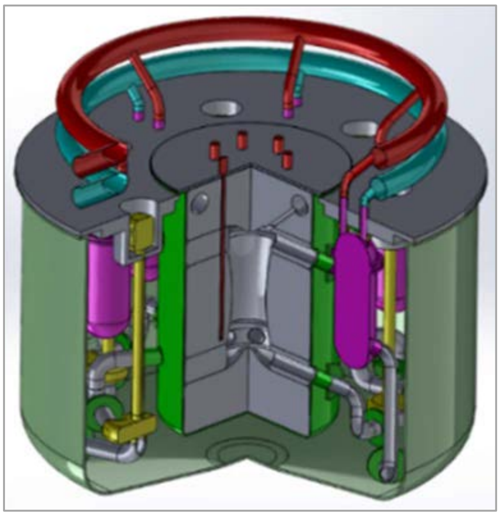

FIG. 1. Schematic view of ARAMIS design (Advanced Reactor for Actinides Management in Salt).

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Paper ID#25

INTEGRATION OF MSR IN LW-MSR FLEETS TO CLOSE THEIR FUEL CYCLE AND/OR MANAGE WASTE

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Abstract

Orano offers the service of recycling U and Pu from LWR operators worldwide, thanks to the La Hague plant (France), a mature industrial facility using a hydrometallurgic process (PUREX). Orano has initiated in 2019 an exploratory programme on fast spectrum Molten Salt Reactors (MSRs) which could be used as efficient convertors of Pu and Minor Actinides (MA). At the same time, Orano would become a fuel salt supplier and a service provider for fuel salt management, enlarging its core business as a fuel cycle leader.

1. INTRODUCTION

Reprocessing of Light Water Reactor (LWR) spent fuels allows to recover 96% of valuable material (Uranium – U - and Plutonium - Pu) still encompassed in the UOX spent fuel, while the remaining 4% are conditioned and safely stored. U and Pu can be recycled in new fuel assemblies for LWRs, reducing the need for natural resources and reducing the radioactive waste disposal footprint as well.

2. DISCUSSION

Today, Orano can offer this service to LWR operators worldwide, thanks to the La Hague plant (France), a mature industrial facility using a hydrometallurgy process (PUREX), having a unique world track record of nearly 40,000 tons of Heavy Metal (tHM) of French and foreign spent fuel reprocessed in the last 50 years. In the case of foreign spent fuels, the ultimate waste is sent back to the country of origin, conditioned in Universal Canisters (UC), a standardized waste form suitable for transport, storage, and final disposal. The recovered Pu is used to manufacture new fuel in the Orano Melox plant (MOX), where almost 3,000 tHM of MOX fuels have been fabricated so far.

These "back-end fundamentals" are the same when talking about oxide/ceramics fuels for Light Water type Small Modular Reactors (LW-SMRs): All the reprocessing and recycling experience developed from long date for existing power reactor fuels by Orano in La Hague and Melox is directly applicable to LW-SMRs.

To go one step further in terms of circular economy, and in complement to the LWR multi-recycling strategy under investigation today in France, which could also be applied to LW-SMRs, Orano has initiated in 2019 an exploratory programme on fast spectrum Molten Salt Reactors (MSRs) which could be used as efficient convertors of Pu and Minor Actinides (MA). Indeed, coupling a fast MSR to a reprocessing facility like the one in La Hague seems one of the most promising ways to reduce the volume and the long-term radiotoxicity of the ultimate nuclear waste, MA today conditioned in vitrified canisters being potentially fully converted into Fission Products (FP) with significantly shorter half-lives.

Doing so, Orano would offer with La Hague an enlarged or improved reprocessing service to LWR operators (including LW-SMRs operators), who would have to handle a smaller number of ultimate residues with a reduced long-term radiotoxicity compared to the current reprocessing situation. This could also be a solution for some LWR operators who can't or don't want to recycle Pu in their LWR fleet, as Pu could be used to feed a fast MSR.

At the same time, Orano would become a fuel salt supplier and a service provider for fuel salt management (e.g., logistics, recycling, conditioning, …), enlarging its core business as a fuel cycle leader.

The chloride salt used in fast MSRs could be compatible with the processes implemented in La Hague, which would be indeed an opportunity to develop synergies with the existing plant for salt synthesis and recycling functions (see Figure 1). The implementation roadmap of such a scheme would be done in two steps:

— Step 1: Production of a Pu salt and transport of used (FP+Pu+MA) salt to La Hague for recycling and waste (FP+MA) vitrification;
— Step 2 (with additional functions for MA management in La Hague): Production of a Pu+MA salt and on-line salt recycling and FP-only salt transportation to La Hague for vitrification.

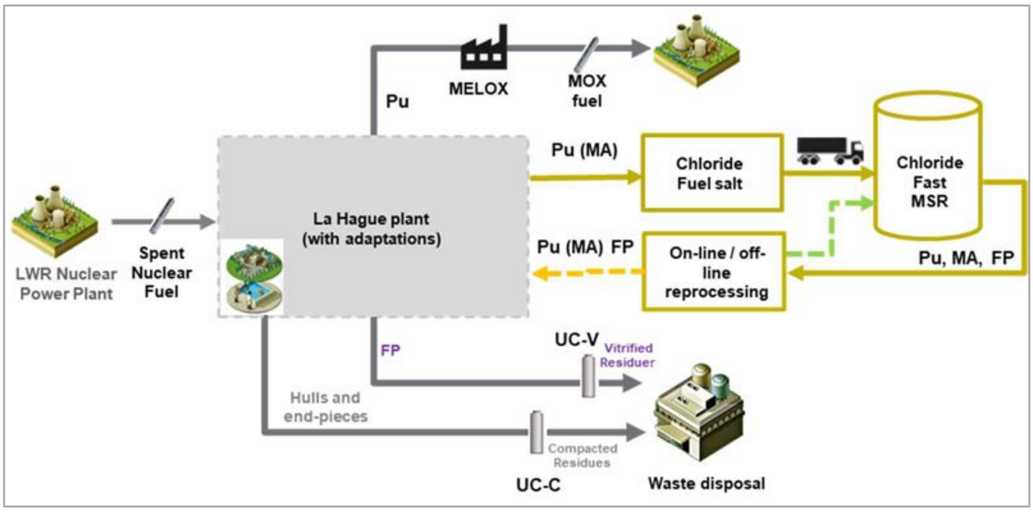


FIG. 1. Coupling the La Hague plant with fast MSRs.

From the viewpoint of LW-SMR operators, several options are possible for their spent fuel management:

— The first one, the open cycle option, consists in conditioning and directly storing spent UOX fuels in a Deep Geological Repository (DGR);
— The second option, or "LWR (mono- or multi-) recycling", focuses on best valorizing the reusable materials that are still encompassed in the used fuel, in LWRs. Mono-recycling corresponds to the actual French fuel cycle strategy. Multi-recycling would consist in recycling in LWRs the Pu contained in used UOX and MOX fuels through a MOX2 fuel. This option could correspond to the middle term French strategy allowing to stabilize the Pu inventory before multi-recycling it in fast reactors;
— The third option, using fast MSRs to convert all remaining Pu isotopes and/or MAs, can be considered as a complementary service to mono- and/or multi-recycling options in LWRs.

A preliminary study has been performed to assess the potential of fast MSRs in a symbiotic fleet of SMRs to "burn" Pu and MA from the LW-SMRs. The methodology was based on the use of several codes and tools for PWR and MSR modeling (see Figure 2):

— CESAR (CEA, Orano) is a simplified depletion code developed for back-end purposes;
— SMURE package (CNRS) is a combination of SERPENT (VTT) and a depletion module;
— AdViCE (Advanced Vitrified Estimator) is an Orano simplified non-industrial tool;

— REM is a code developed at LPSC/CNRS to simulate MSR full core depletion and determine the evolution of MSR salt composition and extracted inventory with different reprocessing choices.

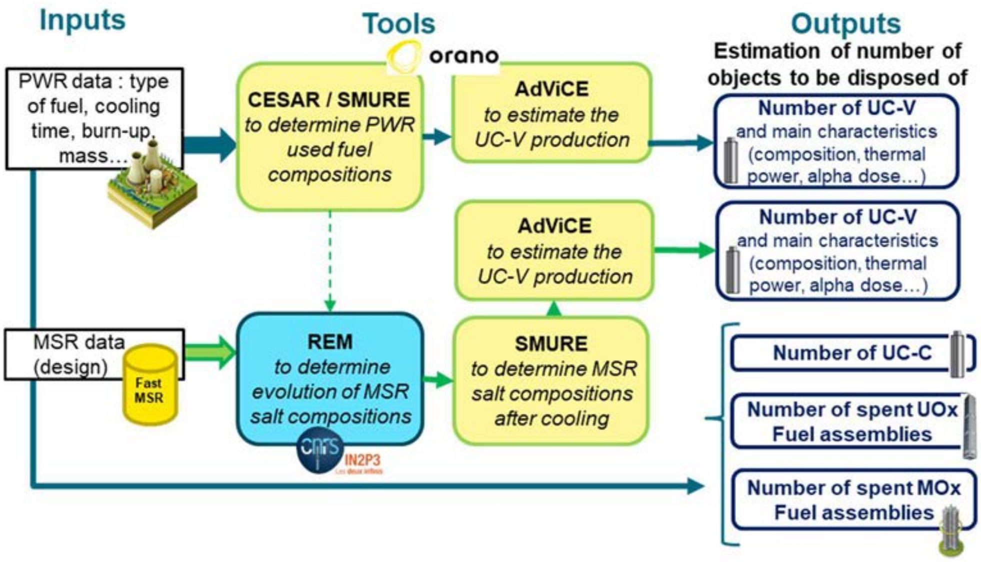


FIG. 2. Methodology used to calculate the number of objects to be disposed of.

The main (simplified) assumptions were the following:

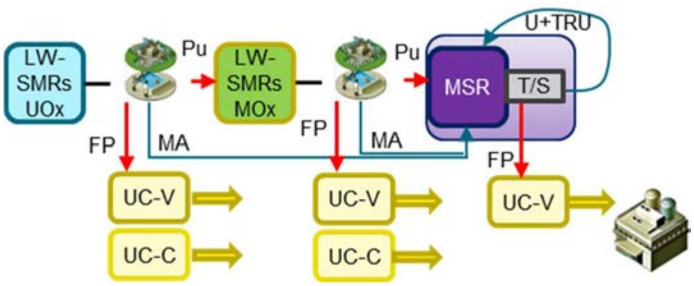
— A fleet producing 10 TWe/y is considered (equivalent to 12 LW-SMRs @ 100MWe);
— LW-SMRs are fictional models, 100 MWe, 13 tHM of fuel in the core and a burnup of 49,5 GWd/t;
— MSRs are fictional models, 135 MWe, a power density of 150 MWth/m3 and a conversion rate (Pu+MA) of ~ 400 g/MWth;
— Steady state compositions are considered for both LW-SMRs and MSRs;
— Reprocessing and vitrification are performed in La Hague (LW-SMRs and MSRs spent fuels): Streams are separated per reactor type, the cooling time is always 5 years before reprocessing, U, Pu and MA extraction rates are considered perfect (100%);
— Reprocessed Uranium to be recycled in LWRs, is not considered in the study;
— Conversion of Pu+MA in MSR will produce used salt with only FP: Pu and MA conversion rates are considered perfect (100%);
— U+TRU are reinjected in the MSR core (on-line ideal & instantaneous reprocessing).

The comparison of the 3 options illustrates the potential of MSRs in reducing the volume and radiotoxicity of High-Level Wastes (HLW). A “symbiotic fleet” of LW-SMRs and MSRs is achieved when actinide production in LW-SMRs equals actinide disappearance in MSRs; this is the optimal scenario for closure of the fuel cycle and waste reduction. Based on the models and assumptions used for the estimation, a symbiotic fleet was obtained @ 81% LW-SMR / 19% MSR, representing for the 10 TWe/y example and the chosen power capacity of reactors: ~10 LW-SMRs + ~2 MSRs. It can be noted that a smaller proportion of MSRs (~6%) would be required if Pu was multi-recycled in LW-SMR instead of mono-recycled. Also, scenarios could accommodate MSRs to be built at a later stage (low maturity today) and/or in other countries.

Finally, a Pu+MA conversion solution respecting the principles of circular economy could be based on the coupling of a “symbiotic” fleet of LW-SMRs and MSRs, and a fuel treatment / separation processing plant like La Hague. Using synergies with the industrial capabilities of La Hague can accelerate the development and deployment of such Back-End solutions for any LWR - including LW-SMR - fuel. The resulting “symbiotic fleets”, composed of LW-SMRs and a limited proportion of MSRs built at a later stage, would provide the owners of such fleets a unique value in terms of sustainability and public acceptance of nuclear energy in the future.

TABLE 1. COMPARISION OF THE 3 BACK-END OPTIONS IN TERMS OF IMPACT ON WASTE TO BE EMPLACED IN A DGR [1]

<table>
<tr><th>Option</th><th>Scheme</th><th>Impact on waste to be emplaced in a DGR</th></tr>
<tr><td>Open cycle</td><td>
</td><td>130 UOX fuel assemblies per year</td></tr>
<tr><td>Mono-recycling of Pu in LW-SMRs (MOX)</td><td>

Pu stream is balanced in MOX fuels</td><td>14 MOX fuel assemblies + 20 UC-V + 23 UC-C per year</td></tr>
<tr><td>Mono-recycling of Pu in LW-SMRs (MOX) + MSR as Pu+MA convertors</td><td>

Pu ex-MOX & MA ex-UOX/MOX streams are balanced in MSRs</td><td>20 UC-V without MA + 21 UC-C per year

No MA in UC-V è strong reduction of both thermal impact and of requirements for the DGR</td></tr>
</table>

ACKNOWLEDGEMENTS

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Paper ID#26

APPLICATION OF A GRADED APPROACH TO THE CONCEPT OF FUEL RECYCLING

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Abstract

Among the categories of SMRs currently being developed, the fast spectrum wasteburner reactors represent the fuel cycle strategies that offer the most straightforward synergies with current fuel cycle and waste management options, with the prospect for a long-term, truly sustainable energy production. Such opportunities in terms of both legacy waste management and fuel availability for the new reactors require the wastes to be recycled. Reprocessing and recycling of used fuel are however considered as sensitive activities in terms of proliferation since early technologies could be suitable for the production of weapons-grade material. With the recent development of commercially available reactors that do not need high purity fuel and recycling technologies that do not involve separation of plutonium from some other elements, proliferation-resistant recycling options can now be considered on an industrial scale. The remaining presence of other species with recycled plutonium brings several independent technical barriers to proliferation, beyond the barriers related to the isotopic composition of plutonium itself. These features of inherent proliferation resistance represent a framework for the application of the concept of graded approach as a basis for a non-proliferation policy towards fuel recycling technologies.

1. INTRODUCTION

With the use of legacy nuclear waste as fuel for further energy production, the fast spectrum Stable Salt Reactor - Wasteburner (SSR-W) represents the cornerstone of a nuclear fuel cycle strategy that offers straightforward synergies with current fuel cycle and waste management options, with the prospect for long-term, truly sustainable energy production.

Such opportunities in terms of both legacy waste management and fuel availability for the new reactors require the wastes to be recycled. Reprocessing and recycling of used fuel are however considered to be sensitive activities in terms of proliferation since early technologies for this could be suitable for the production of weapons-grade material. The SSR-W, however, does not need high purity fuel, and the Waste-to-Stable-Salt (WATSS) technology developed to turn irradiated fuel into fresh fuel does not enable the separation of plutonium from many other elements, in particular most minor actinides and some lanthanides. It is based on a pyrochemical processing technique relying exclusively on chemical redox potentials for the extraction of components of spent fuel. With this kind of separation method, all transuranics display very similar behaviour and the similarities in terms of extraction between higher actinides and lanthanides is highlighted by the fact that cerium is used as a surrogate to plutonium in Pu-free development experiments.

Based on the above considerations, it is therefore important to acknowledge the fundamental differences between this process and other, earlier separation technologies.

2. HISTORICAL BACKGROUND

Weapons-grade fissile material is not readily available and is produced through technological processes that are considered sensitive in terms of export control and non-proliferation safeguards. Uranium highly enriched in U-235 is obtained through isotopic enrichment processes, while plutonium (essentially Pu-239 is useful for military applications) produced in reactors is separated from other elements during “reprocessing” chemical treatments with no change in isotopic ratio from the spent fuel.

While the United States considered in the early years of civil nuclear developments that the recycling of irradiated nuclear fuel in a closed fuel cycle would be a vital contribution towards meeting long-term energy requirements, subsequent US Administrations have pushed for strict controls over enrichment and reprocessing technologies in a bid to limit proliferation of nuclear weapons technologies. This trend culminated under President Carter, who in 1977 issued a strict nuclear policy statement that included banning indefinitely commercial reprocessing and recycling of plutonium in the US nuclear power programme [1]. Such a broad and wide-ranging ban, covering far more than the technologies required to extract plutonium of a military quality, was justified in that country by the fact that the commercially available separation technologies at that time would be suitable to produce purified plutonium, although other countries have successfully contributed to the international efforts regarding nuclear non-proliferation without such a ban. The US ban was also considered to be appropriate in that country because it was considered that a viable and economic programme could be maintained with an open, once through, fuel cycle. Such a conclusion may not be as clear almost a half century later in the perspective of long-term sustainability.

The position of the United States in the framework of nuclear material processing at large is of critical importance since it has largely influenced the international non-proliferation and export-control regime. Although the ban on commercial reprocessing and recycling was lifted by President Reagan, the National Energy Policy issued under President G. W. Bush underscored the necessity to “discourage the accumulation of separated plutonium” [2]. This wording was however elaborated as a need to promote “recycling technologies that do not produce separated plutonium”. Despite these evolutions, most regulatory aspects developed during the 70’s are still in force. The bilateral agreement between the US and Canada, amended in 1980 to reflect the nuclear policy of the Carter Administration [3], has only been marginally amended since then. Efforts to update the “reprocessing regulatory framework” in the Code of Federal Regulations were discontinued in 2021 based on “the estimated costs and the limited interest, expressed or expected, from potential applicants and advanced reactor designers in building facilities involving reprocessing technologies in the near-term” [4].

3. DEFINITIONS

In a gap analysis on the reprocessing regulatory framework from the US Department of Energy [5], one of the major issues was found to be the lack of definitions of the terms “reprocessing” and “recycling”. This issue has not been resolved before the efforts were discontinued in 2021, as mentioned previously.

There are no definitions of either term in Canadian law, but the list of activities mentioned in the “Protocol Additional to the Agreement between Canada and the International Atomic Energy for the Application of Safeguards in connection with the Treaty on the Non-Proliferation of Nuclear Weapons” states that “Reprocessing irradiated nuclear fuel separates plutonium and uranium from intensely radioactive fission products and other transuranic elements” [6]. The same wording is used to define export control requirements in the trigger list of the Nuclear Suppliers Group [7], as well as in an annex of 10 CFR part 110 [8], the only place in the US Code of Federal Regulations where reprocessing is defined. This wording can therefore be seen as a widely adopted definition of reprocessing.

In the “IAEA Safeguards Glossary” [9], reprocessing is not defined as an activity but there is a definition of a “reprocessing plant” as “an installation for the chemical separation of nuclear material from fission products, following dissolution of spent fuel”. The definition goes on to state that "Reprocessing involves purification of uranium and plutonium." The specific inclusion of purification of uranium and plutonium is consistent with the definition in the previous paragraph. In the “IAEA Safety and Security Glossary” [10], the concept of reprocessing is far broader and includes all processes and operations “the purpose of which is to extract radioactive isotopes from spent fuel for further use”, thereby also covering activities intended to recover radionuclides for medical applications.

Besides the general concept for recycling as “The process of converting waste materials into new products” as reported in the section “Minimization of wastes” in [10], authoritative descriptions of the specific features constituting the process of “recycling” in the context of the nuclear fuel cycle do not seem to be available. In a White Paper by the US Nuclear Regulatory Commission intended to propose ways forward in the update of the regulatory framework [11], a footnote mentions that "For the purpose of this document, "recycle" involves (a) separation of the constituents of spent nuclear fuel, (b) refabrication of fresh fuel containing plutonium, minor actinides and possibly some fission products, (c) management of solid, liquid and gaseous wastes and (d) storage of spent fuel and wastes." Such a definition would be suitable for the concept of "fuel recycling" and be in line with the previously reported political will to promote "recycling technologies that do not produce separated plutonium". The activities described are broader in scope than the activities included in the definition of reprocessing but do not involve the separation of minor actinides from plutonium, which is a critical difference in the context of evaluating the implications of a technology from a non-proliferation perspective. Note that the WATSS process not only avoids separation of minor actinides from plutonium, but the process also carries lanthanides along with the plutonium.

Based on the above discussion, we consider that the appropriate definition of "reprocessing" to use for irradiated nuclear fuel is "separating plutonium and uranium from intensely radioactive fission products and other transuranic elements." In addition, we consider that "recycling" of irradiated nuclear fuel should be a process that "involves (a) separation of the constituents of spent nuclear fuel, (b) refabrication of fresh fuel containing plutonium, minor actinides and possibly some fission products" as a minimum – for the purposes of this paper it is not considered that associated waste management and storage activities are relevant to the discussion.

4. TECHNICAL CONSIDERATIONS ON PROLIFERATION ASPECTS OF NON-SEPARATED PLUTONIUM

It is sometimes claimed that even low-quality plutonium could be used to make an explosive device, although of low power and poor reliability. This argument essentially refers to reactor-grade plutonium, in opposition to weapons-grade plutonium. This is chemically pure plutonium but with an isotopic composition that would not be suitable for a nuclear explosive device. In this case, the presence of significant quantities of Pu-240 increases the emission rate of spontaneous neutrons and disrupts the explosive process before the fissile pit is fully compressed. In addition to technical challenges related to the isotopic composition, the remaining presence of other chemical species, especially minor actinides and lanthanides, also brings technical barriers to the production of a nuclear weapon. The relative importance of these barriers to proliferation is the cornerstone of the relevance of the distinctions reported earlier between the concepts of recycling and reprocessing.

The impurities in non-separated plutonium compounds contribute to the emission of spontaneous neutrons, to the heat generation of the material, to its radioactivity and affect the metallurgy of plutonium metal, which is critical to craft a compressible pit. A qualitative analysis of these factors would hardly be carried out since their actual contribution depends on the initial composition of the fuel, the level of burn-up, the neutron energy spectrum in the reactor and the time elapsed since discharge, among other considerations. An overall assessment of their relative contribution would however provide relevant insight.

4.1. Radioactivity

The level of radioactivity of a nuclear material has a considerable impact on its potential diversion towards a weapons programme, due to the constraints related to its handling and transport, the requirements to process it in hot cells and the detrimental effect of radiation on electrical components in a nuclear device. The relative importance of radioactivity in terms of non-diversion is such that this criterium is considered in nuclear security to classify material and determine if it could be categorized as "irradiated", therefore reducing its class of concern [12].

In spent nuclear fuel, the radioactivity is largely dominated by fission products such as I-131 shortly after discharge as well as the lanthanides chain Ce-144 – Pr-144 [13]. After several years, the radioactivity is dominated by isotopes such as Sr-90 and Cs-137, with a significant contribution from Am-241 especially after several decades.

4.2. Spontaneous neutron production

The emission of neutrons from the spontaneous fission of Pu-240 is at the origin of the distinction between weapons-grade (less than 7% Pu-240), fuel-grade (between 7 and 19% Pu-240) and reactor-grade (more than 20% Pu-240) plutonium. When the emission rate is such that the time scale between emissions in a nuclear weapon's pit is lower than the compression time of the pit, these neutrons trigger the pre-ignition of the chain reaction before the critical mass is fully assembled, which then disperses the fissile material. The device is said to fizzle. To our knowledge, all operational Pu-based nuclear weapons have been built with weapons-grade material. One test, in 1962, has been carried out with fuel-grade plutonium, in a bid to assess the technical challenges of using plutonium of sub-optimal isotopic composition [14, 15]. No nuclear explosive device has ever reportedly been built, by any nation, with plutonium having less than 80 % of its 239 isotopes. This highlights that despite claims that a nuclear weapon can theoretically be built with reactor-grade plutonium, the technical difficulties to do it in practice are considerable.

If the plutonium is not chemically pure, other actinides also contribute to the spontaneous production of neutrons. In spent nuclear fuel, the neutron emission is overwhelmingly dominated by the presence of curium, especially the spontaneous fission of Cm-244 and, shortly after discharge, Cm-242 [16]. Several years after discharge, neutrons produced from reactions from Americium 241 also become significant contributors. Overall, the neutron production rate from minor actinides is 2 orders of magnitude higher than the production rate from Pu-240 [16], therefore dwarfing the consideration of isotopic plutonium grade when minor actinides are not separated.

4.3. Heat generation

The issue of heat generation in the fissile material is all but trivial to produce a nuclear weapon. The primary concerns would be to avoid deformation and/or phase changes of the metal pit. In addition to these considerations, an excessive heat would degrade the high-explosives and ancillary equipment required to produce an implosion of a well-defined symmetry required for the ignition.

During the first year after discharge, the most important contributor to heat generation is the lanthanide Pr-144 [13, 16]. The heat generation from the actinide Am-241 is also significant and increases gradually until it becomes the primary source of heat after several decades. In the meantime, it is dominated by Cs-134, Y-90 and Ba-137.

4.4. Alloying behaviour

The crystallographic behaviour of metal plutonium and its alloys is an extremely complex system, and a good control of its phases stability and transition is paramount to enable a suitable compression of the pit. Plutonium is notoriously unstable under almost any external disturbance and exhibits six different allotropic phases with large accompanying volume changes before it melts [17]. Even small amounts of impurities can cause serious degradation of properties because impurities typically concentrate in the melt during casting. This tendency to segregate impurity elements leads to inclusions in the microstructure of plutonium or its alloys [17].

Lanthanide impurities would be of particular concern, from the standpoint of a nuclear weapons developer, because their low solubility in metal actinides would destabilize the alloy. This behaviour is highlighted for example with the phase diagram of the Pu-Ce system which displays no intermetallic compounds [18], in opposition to the behaviour with alloying elements such as Ga or Al [17].

In the presence of other fission products, the behaviour of metal plutonium would also highly likely be considerably altered, although these effects could hardly be predictable due to the variety of compounds making up the fission products in non-separated irradiated fuel.

In plutonium containing minor actinides, the delta phase would be stabilized by the alloying effect of americium [17], which would be interesting for developers of a weapon, although neptunium would tend to destabilize it and form the alpha phase instead.

The relative importance of the different technical considerations discussed above can be summarized visually in Table 1, where the barriers to proliferation are assessed as important, significant or negligible depending on the nature of the remaining elements in the non-purified plutonium compound. These are independent of, and should be added to, the technical issues related to the isotopic composition of the plutonium itself [15].

TABLE 1. RELATIVE IMPORTANCE OF TECHNICAL BARRIERS DUE TO THE REMAINING PRESENCE OF DIFFERENT GROUPS OF ELEMENTS FROM IRRADIATED FUEL FOR THE POTENTIAL USE OF NON-SEPARATED PLUTONIUM COMPUNDS IN THE DEVELOPMENT OF A NUCLEAR WEAPONG. REPORTED IMPACTS ARE ASSESSED AS IMPORTANT (++), SIGNIFICANT (+) OF NEGLIGIBLE (-)

	Radioactivity	**Neutrons production**	**Heat generation**	**Alloying behaviour**
Minor actinides	+ *(1)*	++	+ *(1)*	-
Lanthanides	++ *(2)*	-	+ *(2)*	++
Other fission products	++	-	+	+

**(1) Several decades after discharge / (2) Within first year after discharge*

The considerations developed in this section show that even after separation from most fission products, the remaining presence of minor actinides and/or lanthanides would still carry considerable technical barriers for a military programme. The combination of these technical barriers makes the production of a nuclear weapon with un-purified plutonium theoretically challenging and practically impossible. The relevance of these issues is highlighted by the fact that despite the huge technological and organizational difficulties to produce weapons-grade plutonium, no countries have ever used, or tried to use, lower quality material for military purposes.

These considerations are in line with the concept of “separated plutonium” highlighted earlier as well as a graded approach commensurate on the level of separation of plutonium from other species.

5. CRITICAL ANALYSIS OF THE STUDY ON PROLIFERATION RISK REDUCTION OF “ALTERNATIVE TECHNOLOGIES”

A study released in 2009 assessed the proliferation risk reduction of alternative spent fuel processing technologies [19]. This study has largely supported the ongoing status-quo in the US policy towards nuclear fuel reprocessing. However, if we consider the report in more detail, the implications of the chosen assumptions are as instructive as the findings. For instance, the paper assesses the potential proliferation risk of a technology transfer to a fictional country that already has pre-existing PUREX or equivalent capabilities as well as the resources to build clandestine plants (despite having an Additional Protocol in force, whose purpose is specifically to prevent such undeclared facilities). In sum, they assess the potential for proliferation in a state where the proliferation risk is already elevated due to factors in place before the technology transfer under consideration.

The authors assess that in order to overcome the technical barriers related to the remaining impurities in the product of the alternative separation processes, a further processing with a technology that would actually lead to separated plutonium is required. The ease of conversion of alternative low-temperature liquid-liquid extraction technologies such as COEX or UREX to a full-scope PUREX process is briefly mentioned but molten salt-based equipment and techniques are of a totally different scope. The paper considers that given the capabilities and resources of the state, the further reprocessing with a PUREX-equivalent process could be applied and “there is minimal additional proliferation resistance to be found in introducing alternative processing technologies”. What is not mentioned is that given these assumptions, the state could as well directly divert its irradiated spent fuel in one of its pre-existing covert reprocessing plants. Since the proliferation potential in the different scenarios is largely driven by the resources and capabilities of the country BEFORE any of the potential technology transfers considered, it is obvious that there are only marginal differences between the different end states. Even the absence of technology transfer would lead to a comparable result.

What the study tells us however is that the technical barriers to build a bomb with impure plutonium are so high that the preferred diversion route would require further chemical separation treatment, with another technology. This is the same pathway that would be applied in the absence of performing a process that led to impure plutonium.

If we go beyond the potential ideological or political motivations of the conclusion of the study, a critical analysis of the findings leads to the following implicit conclusions: a) the different barriers to proliferation as discussed in the previous section are indeed considerable, b) the end product of a processing activity that does not lead to purified plutonium still requires a significant conversion time, comparable to that of un-processed irradiated fuel, and c) for a country that does not already have access to proliferation-sensitive capabilities, fuel processing techniques that do not produce separated plutonium do not significantly increase proliferation risks.

6. DISCUSSION ON POTENTIAL DIVERSION OF RECYCLING TECHNOLOGIES

Although there are compelling and well-documented arguments to claim that the use of pyroprocessing in a closed loop cycle can lower the proliferation risk [20], opponents to the technology and more broadly to nuclear technology as a long-term option often refer to a remark from Richard Stratford, an official of the US State Department, at a nuclear policy conference in 2011 to discredit this kind of option. His assertion was that pyroprocessing should be regarded as “reprocessing” on the basis that “electroreduction and electrorefining, the key elements of pyroprocessing, have moved to the point that the product is dangerous from a proliferation point of view”. This claim obviously refers specifically to a type of pyroprocessing based on electrochemical properties in the separation process.

In order to better understand and assess the proliferation concern possibly related to different technologies, it is important to estimate not only the attractiveness of the final product (or lack thereof) but also the potential ease of conversion of a given type of recycling facility. This will be done for four different groups of separation technologies, namely the genuine PUREX process, other water-based separation processes, pyroprocessing using electrochemical separation and pyrochemical processing using exclusively redox potentials. The factors evaluated are the potential direct use of the plant to produce purified plutonium compounds, whether the technological solutions to fill the gap are proven technologies or not, their potential development on the basis of the fielded processes and whether the required equipment for a break-out scenario can be fitted in the existing facility.

The PUREX process represents the baseline in terms of proliferation risk. The technology can be used for the production of purified plutonium and only safeguards provisions and inspections would provide credible assurance that the nuclear material is not diverted. Even if the process is not designed to produce high-quality plutonium, it could easily be optimized for this purpose, or the technology replicated in a clandestine plant.

There are other water (/organic solvent) - based processes such as UREX or COEX that do not produce separated plutonium [19]. However, the technologies required to fill the gap have proved to be efficient, as part of the overall PUREX process developments, and have actually been used for several decades by a few countries. Given the relatively large footprint of such a plant, the systems required to upgrade the facility and convert it into a genuine PUREX plant could be fitted in the existing buildings, once the state has withdrawn or broken out from its non-proliferation obligations.

For pyroprocessing based on electrochemical separation, even though electroreduction and electrorefining processes in general have made significant progresses their specific application to the purification of plutonium still needs to be developed and the suitable parameters optimized. This could only be carried out through dedicated experiments using plutonium, not a surrogate, that would hardly be available without diverting nuclear material. From earlier studies it is assessed that the conversion of a plant based on this kind of technology could be carried out within a few weeks, the high-end of the range for different technologies considered [19]. However, this is assuming that the state has prepared for the breakout by becoming sufficiently familiar with any additional required separations. If such development and familiarization stages require nuclear material to be diverted as a prerequisite, the overall conversion time after breakout would be considerably longer.

In the case of pyrochemical processing relying exclusively on chemical redox potentials, such as WATSS, any improvement of the process would hardly provide fully separated plutonium given the closely related behaviour of higher actinides and some lanthanides. Different separation stages, based on different technologies, should be applied to fill the gap to proliferation. Given the small footprint and difficult physical accessibility of the processing cells, the required separation stages could not be fitted to the existing facilities. In sum, the production of separated plutonium would require another facility, based on another technology and further technological developments, which are requirements driving the potential for proliferation to the same level as the absence of a recycling plant.

TABLE 2. ASSESSMENT OF DIFFERENT FACTORS DRIVING THE RELATIVE EASE OF CONVERSION OF RECYCLING PLANTS BASED ON DIFFERENT SEPARATION TECHNOLOGIES INTO REPROCESSING FACILITIES ABLE TO PRODUCE SEPARATED PLUTONIUM

	PUREX	**Other water / organic solvent extraction processes**	**Pyroprocessing, using electrochemical processes**	**Pyrochemical process, based on chemical redox potential**
Facility readily useable for purified Pu production	X	-	-	-
Technologies to bridge the gap are proven	X	X	-	-
Required technologies could be developed on the basis of recycling facility processes	X	X	(X)	-
Required equipment can be fitted in existing recycling facility	X	X	X	-

The considerations developed in this section highlight a gradation in the potential risk of proliferation, which is a basis for a risk-informed graded approach that would in turn promote the most proliferation-resistant pathways.

7. CONCLUSIONS AND WAYS FORWARD

In a quest to secure long-term availability of energy while also alleviating detrimental effects of climate change, it is expected that a growing number of countries will consider the use of nuclear energy, and especially a closed-cycle nuclear framework. Their legitimate request to recycle their spent fuel and to exploit recycling facilities should be balanced with the potential proliferation risk that it represents. If the International Community fails to acknowledge that different technologies represent different levels of proliferation risk, it would be ill-equipped and would have no leverage to promote the most proliferation-resistant options and prevent the spread of the most proliferating ones.

In this paper, different levels of proliferation concerns have been highlighted in terms of both the potential use of processed material and conversion of recycling facilities. Regarding the proliferation attractiveness of the products of separation technologies, a significant distinction in terms of licensing and regulatory approach can be applied by the adoption of specific definitions for the concepts of “reprocessing” and “recycling”. When it comes to the assessment of the potential ease of misuse of a given technology for the development of a military programme, a further gradation in the potential risk of proliferation has been highlighted, which also forms a basis for a risk-informed graded approach that would in turn promote the most proliferation-resistant pathways.

Such a risk analysis should not be seen in the framework of a stand-alone barrier against proliferation but in a wider defence in depth against proliferation that also accounts for an inspection regime based on the provisions of a Comprehensive Safeguards Agreement and its Additional Protocol, diplomatic efforts to assure that the state remains indefinitely committed to its engagements under the Non-Proliferation Treaty, as well as wider security assurances and protection that remove the rationale for a domestic nuclear weapons programme.

In a comprehensive approach to a risk-informed assessment, it should also be taken into account that the proliferation risk partly depends on the availability of plutonium-containing feedstock, especially the overall stockpile of spent fuel. The removal of actinides from this stockpile, combined with suitable transmutation that fast spectrum SMRs can offer, represents a risk-reduction trend that needs to be factored in any assessment of the proliferation impact of the future adoption of a recycling technology.

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Paper ID#33

CHALLENGES OF SMALL MODULAR REACTOR USED FUEL MANAGEMENT IN CANADA

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Extended Abstract

Currently, Canada is considering the deployment of three streams of small modular reactor (SMR) technologies in different province: 1) grid-scale SMRs based on light water reactor technology (LWR-SMR), such as boiling water reactors (BWRs), 2) SMRs based on sodium-cooled fast reactors (SFR-SMR), and molten-salt fuel reactors (MSR-SMR), and 3) micro-SMRs based on high-temperature gas-cooled reactors (HTGR-SMR). Other reactor technologies for SMR applications are also under considerations. Due to their small power generation capacities (15 MWe to 300 MWe) and the anticipated deployment to dozens or hundreds of small, remote, and unconnected communities, it is expected that a large number of units will be required. It is also anticipated that the most economical used fuel management approach for a large fleet of SMRs will require the use of centralized facilities. This paper provides a brief review of challenges and opportunities that may be presented by SMR used to current used fuel management in Canada: wet storage, dry storage, direct geological disposal, and future processing and recycling of partially used nuclear fuels.

By using enriched and higher-fissile-content fuels, SMRs are expected to achieve much higher levels of burnup than Canada Deuterium Uranium (CANDU®) pressure tube heavy water reactors (PT-HWRs) with natural uranium fuel, and thus will have a reduced mass and volume of spent fuel per unit of electrical energy generated. However, a key trade-off is that the high burnup levels in SMR fuels will lead to greater inventories of radioactive activation products, fission products, and minor actinides in used fuel, and thus higher levels of gamma, neutron, radiotoxicity, and decay heat generation, relative to conventional used CANDU®. These higher radionuclide inventories and radiation will impact the used fuel storage, transportation, and disposal or processing and recycling. The greater or different complexity of the fuel designs for the BWR-SMR, HTGR, SFR and MSR technologies will create further challenges for used fuel management. Such challenges may include: 1) the cooling of metallic, molten salt, and TRISO fuels immediately after discharge as they are unlikely suitable for wet storage, 2) conditioning requirements for their disposal with the CANDU® fuel, and 3) solutions for transportation of used fuel from remote communities to a centralized facility.

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OVERVIEW OF THE U.S. DOE'S OFFICE OF SPENT FUEL & WASTE DISPOSITION ACTIVITIES

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Extended Abstract

In the United States, nuclear energy is essential to tackle climate change. It currently provides 20 percent of the U.S. electricity and half of the U.S. clean, emission-free energy [1]. As the U.S. continues to deploy nuclear energy as a solution for decarbonization, further progress needs to be made on the management of spent nuclear fuel. Final management of the Nation's spent nuclear fuel and high-level radioactive waste is the U.S. Department of Energy's (DOE) responsibility. The DOE's Office of Nuclear Energy (NE) is responsible for ongoing research and development (R&D) related to long-term disposition of spent nuclear fuel and high-level radioactive waste, which are managed by the Office of Spent Fuel and Waste Disposition (SFWD). SFWD has two offices that cover different aspects of this oversight: the Office of Spent Fuel & Waste Science and Technology (SFWST) and the Office of Integrated Waste Management (IWM). The SFWST office has developed and is executing an R&D programme that will address critical scientific and technical issues associated with the long-term management of spent nuclear fuel. The IWM office supports evaluations, planning, and preparations for transport and disposal of spent nuclear fuel and high-level radioactive waste and the possibility of interim storage for spent nuclear fuel. Activities include conducting system analyses to evaluate the integrated approach for transport, storage, and disposal; identifying potential functional and operational requirements; examining interim storage design configurations; planning for transportation of radioactive materials with appropriate stakeholder interactions (including states and tribes); prototype railcar development and testing; and routing analysis tool development.

While spent nuclear fuel is stored safely all over the U.S., the communities where spent nuclear fuel is located have never agreed to host this material in the long term. This is a problem that should not be deferred to future generations to solve. As part of this effort, DOE is working on an integrated waste management plan that will include federal consolidated storage capability, transportation infrastructure, and a pathway to a disposal. The following paragraphs list some of the activities currently pursued by the SFWD office.

[1] Federal Consolidated Interim Storage Capability and Consent-Based Siting

In the Consolidated Appropriations Act 2021 [2], U.S. Congress provided funds for DOE to move forward with establishing a federal interim storage capability. In 2022, Congress continued to fund this activity. DOE is using consent-based siting to identify willing and informed host communities to host one or more federal consolidated interim storage facilities. Interim storage will allow for the removal of spent nuclear fuel from reactor sites, promote new jobs and economic opportunities for host communities. It will also provide useful research opportunities and build trust and confidence with stakeholders, communities, and the public by demonstrating a consent-based approach to siting.

Consent-based siting is an approach to siting that makes the needs of people and communities central to the process. In this process, communities can elect to participate by working collaboratively through a series of steps and phases with DOE as the implementing organization. Each step and phase of the process helps a community determine how hosting a facility aligns with the community's goals. The

process is not intended to serve as a guide, or a prescriptive set of instructions. Consent-based siting will look different for each individual interested community. Some potential outcomes from this process could include: a negotiated consent agreement (as defined by the community in collaboration with DOE), or a determination that after exploring the process, the community is simply not interested. DOE considers both outcomes to be a success.

On December 1, 2021, DOE issued a Request for Information (RFI) on using a consent-based siting process to identify sites to store the nation's spent nuclear fuel [3]. The RFI was open for comments for three months until early March 2022, its purpose was to hear from the public on the following topics:

— The consent-based siting process itself;
— Removing barriers to meaningful participation—especially for groups and communities who have not historically been well-represented in these conversations;
— Interim storage as a component of the nation's waste management system.

DOE is ensuring issues of equity and environmental justice are built into the consent-based siting process. A total of 225 responses were received from Tribal, State, and local governments and organizations, as well as from a broad range of stakeholders including non-governmental organizations, industry, members of academia, and private citizens. Comments in their entirety and a summary of the comments' analysis are available on the DOE website [4]. DOE has been using the feedback it received to inform the next steps, this includes the recently issued $16 million funding opportunity announcement (FOA) to provide resources for communities interested in learning more about consent-based siting, management of spent nuclear fuel, and interim storage facility siting considerations [5]. Award recipients will help to promote dialogue that fosters the development of innovative community ideas, incorporates principles of equity and environmental justice into community engagement strategies, and captures feedback related to the interim storage of spent nuclear fuel and high-level radioactive waste. Tasks supported by the funding are divided into the three following areas:

— Organize, lead, and maintain meaningful, inclusive community and stakeholder engagement processes related to management of spent nuclear fuel;
— Elicit and map public values, interests, and goals to promote effective collaboration and community-driven insights and feedback on a consent-based siting process for a federal consolidated interim storage facility;
— Develop, implement, and report outcomes and strategies that support mutual learning among stakeholders, communities, and experts on nuclear waste-related topics.

[2] Advanced Reactor Waste Management

The SFWD office currently leads a team of both DOE federal staff and technical staff from U.S. National Laboratories engaged in the characterization of the fresh fuel, as well as the spent nuclear fuel that would result from the operation of the different advanced reactor designs being proposed. There are ongoing efforts to develop an integrated high-level strategy to identify additional waste management issues for advanced reactor spent nuclear fuel and accident tolerant fuel from existing and future fuel cycles. TRISO fuels and metallic fuels are a primary focus of these efforts.

There is ongoing work aimed at developing a preliminary framework to help DOE plan for the deployment of facilities that manage advanced reactor spent nuclear fuel in the direct-disposal fuel cycle. DOE is also working to identify the advanced reactor waste types that may require treatment prior to disposal. This could help inform where technical and regulatory gaps exist. This could be a topic that multiple countries could benefit from collaboration on.

[3] Transportation, Storage, and Systems Analysis

There is research on extended storage of spent nuclear fuel, aging management, and subsequent transportation with the purpose of closing knowledge gaps and improving confidence through experimental testing and model validation. DOE is developing purpose-built railcars for future large-scale DOE transport of spent nuclear fuel from nuclear power plants, planning to conduct full-scale accident testing of a rail-sized spent nuclear fuel transportation cask, evaluating transportation infrastructure conditions and potential refurbishment needs to support eventual removal of spent nuclear fuel packages ranging from 82 short tons (74 metric tons) to 210 short tons (190 metric tons), assessing spent nuclear fuel transportation security considerations, and coordinating transportation operational planning and training needs with representatives from Tribal and State governments through whose jurisdictions DOE may transport spent nuclear fuel to storage and disposal facilities.

To prepare for implementation of a federal consolidated interim storage capability, DOE is developing conceptual storage facility designs, preparing technical documentation to support project management, analyzing regulatory considerations and approaches to facility licensing and aging management plans, and is demonstrating options for conducting receipt inspections of spent nuclear fuel canisters.

Finally, systems analysis work connects the components of a waste management system. DOE is developing and refining computational tools to model waste system throughputs, schedules, equipment needs, and interfaces to support decision-making and strategic planning for US nuclear waste management. Together, each of these programme areas support DOE's implementation of a waste management system including consolidated interim storage, transportation, and eventual disposal of spent nuclear fuel and high-level radioactive waste.

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Paper ID#35

PROGRESS ON CONSIDERING THE BACK END OF THE FUEL CYCLE FOR SMALL MODULAR REACTORS

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Abstract

Several small modular reactor (SMR) design concepts are under development in the United States and around the world. As SMRs are developed and begin the Nuclear Regulatory Commission (NRC) licensing process in the United States, attention to the back end of the fuel cycle needs to be considered. The use of SMRs presents a challenge to the backend of the nuclear fuel cycle, as storage, transport, and disposal are designed, regulated, and built around traditional LWR spent fuel. A broad review of publicly available SMR information to get a sense of back-end fuel cycle differences or gaps that should be considered for SMR SNF transportation and storage was conducted. Storage and transport requirements are described in Federal Regulations, that require specific packaging performance for normal transport conditions (NCTs) and assumed accident conditions (HACs) where the packaging needs to survive without loss of shielding, critical and containment safety functions. Also, they stipulate licensing requirements, procedures, and criteria for receiving, transferring, and retaining other radioactive materials related to SNF storage. The Nuclear Regulatory Commission may decide to revise these nuclear regulations and guidelines to suit most SMR waste management systems. Overall, the conceptual handling of SMR SNF can borrow heavily from what is currently known for LWR SNF. Therefore, SNF management vendors will need to develop increasingly more detailed system designs and work closely with reactor vendors who in turn will need to coordinate with the NRC and DOE to understand the impact of their SNF on established and future management systems.

1. INTRODUCTION

The Nuclear Waste Policy Act of 1982, as amended, established the federal government's responsibility to accept Spent Nuclear Fuel (SNF) and High-Level Radioactive Waste (HLW) from waste owners and generators for disposition. Most envisioned SNF was fuel assemblies from LWRs. This SNF is well characterized and currently being stored at reactor sites in water-cooled pools or dry air-cooled Independent Spent Fuel Storage Installations (ISFSI).

SMR design concepts are under development in the United States of America and across the world. As SMRs are developed and begin the NRC licensing process in the United States, attention to the back end of the fuel cycle needs to be considered. These design concepts are envisioning the use of molten salt, TRISO, metallic, and other novel fuel forms. The use of these reactors presents a challenge to the back end of the nuclear fuel cycle because storage, transportation, and disposal has been designed, regulated, and built around established LWR fuel.

This extended abstract summarizes the conclusions from our work performed this year for the Office of Integrated Waste Management within the Department of Energy of the U.S. federal government.

2. REPRESENTATIVE FUELS

We conducted a broad review of publicly available SMR information to get a sense of back-end fuel cycle differences or gaps that should be considered for SMR SNF transportation and storage.

— *Advanced LWR*: Near-term SMRs use shorter versions of traditional Boiling and Pressurized Water Reactor fuels characteristic of the existing fleet. As with longer LWR fuel, when the SMR SNF is removed from the core, it will be stored in a spent fuel pool and eventually transferred to dry storage casks as needed;

— *Salt-fueled Molten Salt Reactors (MSRs)*: When removed from the core, the MSR fuel salt can be heavily poisoned and pumped into shielded subcritical geometry decay canisters. As the canisters cool, the fuel salt will solidify. Online polishing could provide the potential for the MSR SNF to be classified as HLW. Furthermore, post shutdown salt treatment and separation activities may allow a portion of the MSR fuel salt to be classified as low-level radioactive waste;

— *High Temperature Gas-cooled Reactors (HTGRs)*: The TRISO fuel is in the form of pebbles or prismatic blocks. A modified canister design will likely be required for TRISO fuel pebbles. Dispersal of boron-doped spheres among the TRISO fuel pebbles to replace the current dry storage basket grids used for LWR fuel may also be necessary. Consideration for safeguards, subcriticality, shielding, and cooling of the very large number of TRISO fuel pellets will be necessary with two likely options: direct disposal of the graphite fuel element or separation of the fuel pellets from the larger graphite block;

— *Fluoride Salt-cooled High-Temperature Reactor (FHR)*: Considerations for the FHR SNF are similar to those for the HTGR SNF except the TRISO fuel pebbles may need washing before placing them in a dry storage canister. The final storage option of this graphite fuel element may differ if the FHR uses prismatic fuel;

— *Sodium Cooled Fast Reactors (SFRs)*: Some plans use sodium-bonded uranium-zirconium (U-Zr) metallic fuel clad in HT9 stainless steel. This fuel form and cladding material have a long history of research and development and were demonstrated in the Experimental Breeder Reactor - II. Sodium-bonded fuel may need to be processed to remove the bonded sodium prior to disposal because it is generally considered unsuitable for direct disposal. An alternative SFR fuel is annular, sodium-free, binary U-Zr metallic fuel clad in HT9. This advanced fuel design is expected to enable the high burnup required for breed-and-burn operation;

— *Lead-cooled Fast Reactors (LFRs)*: LFRs will likely use uranium and plutonium dioxide, or nitride ceramic fuel pellets with < 20% enrichment inserted into stainless steel clad and backfilled with helium. The fuel pins are would then be inserted into square or hexagonal fuel assemblies.

3. REGULATORY GAPS

The storage and transportation requirements are delineated in the Code of Federal Regulations (CFRs) 10 CFR 71 and 10 CFR 72. Regulations in 10 CFR 71 require specific package performance for Normal Conditions of Transport (NCT) and postulated Hypothetical Accident Conditions (HACs) that the package needs to survive without loss of its shielding, criticality, and containment safety functions. The regulations in 10 CFR 72 establish requirements, procedures, and criteria for the issuance of licenses to receive, transfer, and possess SNF other radioactive materials associated with its storage. Various regulatory guides and regulations provide information for vendors and utilities to meet 10 CFR 71 and 10 CFR 72 requirements for LWR SNF.

NUREG-2215, Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities, provides NRC staff guidance to review the safety analysis report associated with:

— A certificate of compliance for a dry storage system for use at a nuclear power reactor;
— A specific license for a dry storage facility.

The NRC review evaluates the proposed dry storage system or facility design, contents, operations, and the proposed site to ensure the application provides reasonable assurance that the design and operations meet the regulations in 10 CFR 72. The review is expected to “identify and resolve potential design or operational deficiencies, analytical errors, significant uncertainties, or non-conservatisms in design approaches. Items evaluated include the uniqueness of the design (as compared to existing designs), safety margins, operational experience, defence in depth, and the relative risks that have been identified for normal operations and potential off-normal conditions (or anticipated occurrences) and accident conditions. References in NUREG-2215 are generally related to LWR fuel” [1].

RG 3.54, Spent Fuel Heat Generation in an Independent Spent Fuel Storage Installation, provides a methodology that is appropriate for computing the LWR fuel assembly heat generation rates as a function of burnup, specific power, decay time, and enrichment up to 5 wt% uranium-235. The target limits of this regulatory guide include fuel with enrichments up to 5 wt% and burnup limits of 55 GWd/MTU for BWR fuel and 65 GWd/MTU for PWR fuel [1].

NUREG-2216, Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material, provides guidance for reviewing applications for approval of package designs used for the transport of radioactive materials under 10 CFR 71. Certificate holders are responsible for demonstrating that the package design meets the requirements and licensees are responsible for complying with the general license. Several considerations may influence the depth and rigor that is needed for a reasonable assurance determination of both safety and compliance including the novelty of the design (as compared to existing designs), safety margins, operational experience, and defence-in-depth [1]. The experience base is with LWR fuel.

RG 3.71, Nuclear Criticality Safety Standards for Nuclear Materials Outside Reactor Cores, describes methods that are acceptable for criticality safety standards associated with nuclear materials outside reactor cores, including procedures for preventing nuclear criticality accidents in operations that involve handling, processing, storing, or transporting special nuclear materials or a combination of these activities.

The Nuclear Regulatory Commission may decide to revise these nuclear regulations and guides to adapt them for most SMR waste management systems. Identifying specific thermal, shielding, material, dose, criticality safety, and other regulatory gaps was beyond the scope of our preliminary analysis. The burden will fall on vendors and utility owners-operators to make the regulatory case for onsite storage and preparation of their fuel forms for transportation. However, from a high-level review of the SMR technologies, fuel forms, storage requirements, and transportation requirements, there are no obvious roadblocks to handling SMR SNF in the back end of the fuel cycle while meeting the requirements laid out in 10 CFR 71 and 10 CFR 72.

4. RADIOACTIVE WASTE MANAGEMENT

Treatment involves the conversion of radioactive wastes into new forms. For SMR SNFs, treatment may be necessary to ensure they meet transportation, storage, or disposal requirements. Alternatively, treatment may be elective if the treatment simplifies or streamlines subsequent SNF management operations. Treatment is conceptually distinct from reprocessing because it does not entail the intentional separation of fissile, fertile, or fission-product species in the SNF.

4.1. SNF Conditioning and/or Treatment Not Necessary

Waste treatment is not required for SNF from SMRs based on current LWR fuel technology, which has previously been assessed to be an acceptable waste form for transportation, storage, and final disposal in a geologic repository. Fuel pebbles, such as those used in some types of HTGRs and salt cooled MSRs, are made of graphite with TRISO fuel particles embedded in the inner region of the spherical pebble. The pebbles themselves are expected to be a suitable waste form for transportation, storage, and disposal based on their chemical durability, physical resilience, and thermal robustness without any further waste treatment or conditioning.

No waste treatment is foreseen for annular metal fuel without sodium. The treatment ultimately depends on the materials used in the sodium-free bond, but such fuel is currently expected to be suitable for transportation, storage, and disposal.

No waste treatment is foreseen for the reviewed LFR fuel designs. These are backfilled with helium and thus should pose no issue for management and disposal. Alternatives backfill materials, such as lead, may be assessed as more problematic and could require treatment.

4.2. SNF Conditioning and/or Treatment Necessary or Elective

Waste treatment of the fuel salt from a salt fuelled MSR will depend on the specific reactor type, online salt conditioning, and any reprocessing steps taken during reactor operation. Elective post-operation treatment of the fuel salt may reduce waste volume and allow the recovery of expensive salt or other materials for reuse. If destined for final disposal, however, all once-through spent liquid fuel salts may require treatment prior to disposal in a geologic repository because they are soluble in water. The one exception is for direct disposal in a salt repository because the presence of subsurface salt beds indicates a long-term absence of water.

Treatment will likely be required to remove sodium metal from sodium-bonded metallic fuels in SFRs. During reactor operation, the sodium bond migrates into the fuel. Sodium metal is reactive with water and therefore is generally assumed to be unsuitable for final disposal in a repository.

Prismatic-block-type fuel, such as that used in some types of HTGRs and salt-cooled MSRs, is generally a solid block of graphite through which fuel and coolant channels run. The fuel channels are filled with graphite compacts in which TRISO particles are embedded. Waste treatment of prismatic-block-type fuel will likely not be required. However, treatment may be elected to reduce the SNF or HLW volume by separating the fuel compacts from the graphite blocks to allow the latter to be managed as low-level waste.

5. STRATEGIC RECOMMENTATIONS

Current work in SMR SNF management is limited to being conceptual due to the lack of operating SMRs. Overall, the conceptual handling of SMR SNF can borrow heavily from what is currently known for LWR SNF. Therefore, SNF management vendors will need to develop increasingly more detailed system designs and work closely with reactor vendors who in turn will need to coordinate with the NRC and DOE to understand the impact of their SNF on established and future management systems.

If vendors and regulators begin working together early enough in the process, the former can ensure regulatory compliance while the latter evolve practices and requirements for such SMR SNF. The following are several areas where this early partnership may prove especially valuable:

— Design of SNF pools or cooling areas;
— Siting and design of onsite SNF storage installations (if applicable);
— Updated Normal Conditions of Transport characteristics for a respective fuel type;
— Assessment of off normal and accident conditions applicable for each fuel type;
— Assessment of bounding and nominal disposal conditions for each fuel type.

Some research and development may be needed if significant technical or regulatory gaps arise concerning the management of SMR SNF when compared to LWR SNF. Therefore, periodic studies of the regulatory and industrial environment of SMRs as the landscape changes will help to identify such gaps including those which national laboratories with extensive SNF-focused research could help to address.

Currently, SMR development is in a state of flux with many vendors pursuing many reactor concepts. Continuing work interfacing with reactor vendors is recommended to understand their plans and how they are changing. This work would ensure that the proposed SMRs' impacts to the back end of the fuel cycle back are well understood and acted upon by SNF stakeholders, well in advance of any SNF being generated.

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Paper ID#36

RECENT INVESTIGATIONS INTO TECHNICAL ISSUES OF THE BACK END OF THE FUEL CYCLE OF ADVANCED REACTOR FUELS AND ACCIDENT TOLERANT FUELS

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Extended Abstract

In the United States, the emergence of accident tolerant fuel (ATF) and advanced reactor (AR) fuel technologies has prompted the U.S. Department of Energy (DOE), the U.S. Nuclear Regulatory Commission, and the commercial nuclear power industry to investigate technical issues associated with the back end of the nuclear fuel cycle (BENFC) for these fuel types. Efforts have been focused on implementing expertise from these institutions in the storage, transportation, and disposal of spent nuclear fuel (SNF) including both AR SNF and ATF using precedents established by handling and evaluation of the current inventory of U.S. SNF. This work highlights the strategy and relevant R&D within the DOE Spent Fuel and Waste Science and Technology (SFWST) Campaign. Because small modular reactor (SMR) concepts range from integral light-water reactors to more advanced reactors with lower technology readiness levels, these approaches can be applied to SMRs as well.

The SFWST Campaign provided a high-level gap analysis for the storage and transportation (S&T) of accident tolerant and advanced fuels [1]. The analysis focused on proposed ATF options currently being irradiated in commercial reactors, since these options are most likely to see more widespread use. AR fuels likely to be implemented were also considered. The cladding technologies investigated were chromium-coated zirconium-based alloys, FeCrAl, and silicide carbide. The fuel technologies investigated were chromium-doped uranium dioxide fuel, uranium alloys, uranium nitride, and uranium silicide. The report acknowledged that institutions in several countries are conducting significant testing and modelling of these technologies to establish initial performance metrics and that additional metrics relevant to storage and transportation will need to be investigated once the ATF and AR fuels have reached end-of-life (EOL) burnup. The primary conclusions from this gap analysis were (1) the current storage and transportation canister/cask systems are capable of handling the rod/assembly ATF systems that are similar to existing commercial spent fuel, but modifications to accommodate increased weight and greater thermal and ionizing radiation loads may be required depending on the fuel; (2) significantly different fuel systems, such as tri-isotropic fuels and uranium alloys, will require new storage, transportation, and disposal container designs.

In addition, in 2020 the SFWST Campaign worked with the DOE's Advanced Research Projects Agency to identify opportunities for reducing the disposal impact of waste generated by AR fuel cycles. A range of AR fuel cycles and fuel forms was considered in the study, and waste form characteristics and requirements that can be used to define "disposal impact" were identified and documented. Waste form characteristics that are important to disposal include the chemical form of the waste; its expected lifetime once the waste package fails; the radionuclide inventory of the waste, particularly of the long-lived fission and activation products and of fissile radionuclides; thermal characteristics; physical characteristics; how the waste is packaged; whether it is vigorously reactive to water; whether it generates gas; and the types of safeguards and security issues associated with the waste [2].

The DOE SFWST Campaign includes plans for testing of rods representative of EOL burnup of ATF and AR fuels [1]. This testing will be conducted using the same (or similar depending on fuel form) processes and procedures that were used in testing the high burnup light water reactor Sibling Pins (fuel rods) from the collaborative Electric Power Research Institute and DOE High Burnup Demonstration project [3]. The post-irradiation characterization serves as an example of the general approach to be taken with ATF and AR fuels. Direct performance comparisons with the majority of the high burnup rods from the current spent nuclear fuel inventory will be made; from there, a technical basis will be established to determine if any accommodations need to be made for spent ATF and AR fuel storage, transportation, and disposal. Parameters of interest include ATF and AR fuel particulate size and quantity; cladding coating robustness and potential corrosion as well as hydride formation in areas of damaged cladding coatings; and increased container weight, temperatures, and radiation levels.

More recently, the SFWST Campaign initiated development of an integrated high-level strategy to identify additional storage, transportation, and disposal issues for AR SNF and ATF from existing and future fuel cycles, as well as to delineate the R&D that would address and close those gaps. The high-level strategy will provide an overview of AR SNF (and other potential waste streams), with a primary focus on TRISO fuels and metallic fuels, but also including coverage of waste from molten salt reactors, small modular reactors, and ATF. Some of the AR SNF will be High Assay Low Enriched Uranium fuels, which are fuels with uranium enrichment above 5 percent and less than 20 percent. The approach will include a survey and description of existing and projected waste forms and existing SNF with defined disposition pathways based on previous gap and safety analyses, including details from existing storage and transportation activities, disposal options assessment, and the regulatory framework across all aspects of the BENFC. Precedents will be mapped to projected waste streams from ARs to use as the basis for an initial BENFC disposition strategy, as well as a reference point from which to identify further gaps to be addressed with additional R&D. In this way, the exercise will assess feasibility of disposition pathways for AR SNF and other potential waste streams. This strategy development will guide detailed gap analyses for S&T and disposal options. These detailed gap analyses will identify and evaluate potential technical gaps in the characteristics, packaging, regulatory, and concepts of operation for AR waste streams, which in turn will delineate further R&D to close those gaps.

Addressing technical issues for the BENFC for ATF and AR SNF, including fuels considered part of SMR concepts, has been an active research area for the SFWST Campaign. Recent and ongoing activities in the SFWST Campaign for high-level gap analyses for storage and transportation of ATF and AR fuels and developing an integrated high-level strategy for identifying and evaluating storage, transportation, and disposal issues for AR SNF (and other waste streams) have been summarized.

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EVALUATION OF ADVANCED REACTOR SPENT FUEL MANAGEMENT FACILITY DEPLOYMENT

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Abstract

Many proposed small modular reactor (SMR) designs are based on advanced reactor system concepts and fuel technologies which differ from those used in light water reactors (LWRs) in operation today. Thus, SMRs may produce spent nuclear fuel (SNF) with characteristics very different than that from LWRs. New planning efforts are needed to anticipate the unique requirements and the challenges on deploying a nuclear fuel cycle back-end management system which can accommodate SNF from advanced reactors. Building on past efforts that focused on LWR SNF management, this paper applies a framework of high-level facility deployment milestones to a generic SNF management system and considers them in concert with the unique characteristics of advanced reactor SNF. As a result, consideration of SNF management challenges is considered with respect to factors that influence facility and system deployment. Four advanced reactor system technology types – sodium-cooled fast reactors, high-temperature gas-cooled reactors, liquid-fuel molten salt reactors, and lead-cooled fast reactors – are considered. Additionally, the unique aspects of microreactor SNF management due to reactor size, portability, and siting are explored. Ultimately, milestones earlier in the facility deployment process (e.g., siting and design) appear to be more impacted by SNF characteristics than others (e.g., construction and testing). In many cases, addressing advanced reactor SNF management challenges could require U.S. Nuclear Regulatory Commission (NRC) rulemaking, Congressional legislation, and/or new considerations for physical security. Ultimately, none of the challenges are seen as disqualifying but they should be considered early to avoid issues in the future.

1. INTRODUCTION

Many proposed designs for small modular reactors (SMRs) are advanced reactors which differ from existing light water reactor (LWR) system designs and technologies. These advanced reactor concepts offer various operational advantages, such as higher thermal efficiencies, improved fuel utilization, and passive safety, among others. Each may produce spent nuclear fuel (SNF) with characteristics very different than the SNF that has been produced to date by LWRs.

Past research and planning for LWR SNF management facility deployment were based on organizing milestones and activities. Activities are any research, development, design, or decision taken in the development and deployment process. Milestones are activity endpoints and comprise the set of achievements required to reach facility deployment. Milestones and activities are arranged into success precedence diagrams to catalog where the implementing organization had (or did not have) full control over milestone success, understand the critical path of milestones on which success was conditional, and quantify timelines, uncertainties, costs, and risks [1-4].

This paper documents preliminary applications of the milestone and activity framework to managing SNF from advanced reactors [5]. The challenging aspects of managing different types of SNF are considered alongside: (a) the types of management facilities required; and (b) high-level milestones applicable to the deployment of any facility. Five different classifications of advanced reactors are considered based on the type of technology or size: sodium-cooled fast reactors, high-temperature gas-cooled reactors; liquid-fuel molten salt reactors; lead-cooled fast reactors; and microreactors.

2. GENERIC SNF MANAGEMENT FACILITIES AND MILESTONES

The management of SNF from any type of reactor in a direct disposal fuel cycle may require facilities to carry out some or all of the following operations:

— On-site cooling and storage after discharge;
— Fuel treatment, if necessary, to remove reactive/hazardous materials or prepare wastes for disposal;
— Off-site storage at a consolidated interim storage facility;
— Final disposal in a geologic repository;
— Transportation between some or all of these steps.

Different advanced reactor types may not require all generic operations, and facilities would need to be tailored to specific types of fuels or processes. Figure 1 shows a generalized, linear flow sheet of these operations in a hypothetical back-end system. Transportation would be required between some, but perhaps not all, steps. A fuel treatment step could come before and/or after interim storage, depending on the purpose(s), e.g., to prepare the SNF for transport, interim storage, and/or disposal.

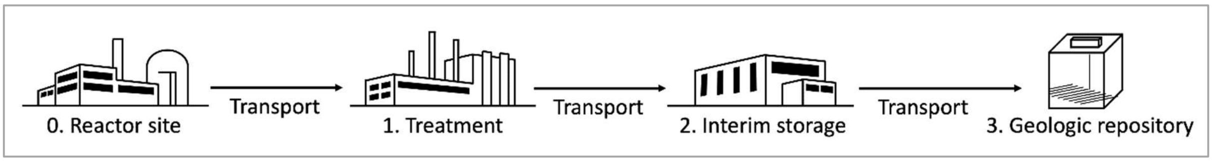

FIG. 1. Generic advanced reactor SNF management flow sheet for direct disposal fuel cycle (no reprocessing), assuming fuel treatment step and transportation between all facilities (reproduced from Ref. [5,6] with permission).

Every facility deployed to carry out one of the aforementioned operations will require achievement of the same set of seven high-level major milestones. These are documented in Table 1. Despite the limited number of milestones, the activities to achieve them are complex and highly coupled.

TABLE 1. HIGH-LEVEL MILESTONES TO DEVELOP SNF MANAGEMENT FACILITIES

Milestone	Activities required
Establishing responsibility	Authorize implementing organization Determine management structure, financial responsibility
Siting	Develop siting process; request/evaluate volunteer sites Negotiate consent agreements; designate site
Establishing transportation infrastructure	Design and test rolling stock; manufacture/procure fleet Design and obtain NRC approval of transportation packages Identify maintenance needs, design necessary facilities Select routes; obtain approvals from authorities; enter necessary contracts
Design	Establish scope, design, acceptance criteria; complete safety analysis
Licensing	Prepare and submit license application to Nuclear Regulatory Commission (NRC), including safety analysis and environmental reports NRC reviews application and prepares environmental impact statement. Obtain necessary licenses/permits from local, tribal, state, and federal authorities
Construction	Prepare site Construct main and ancillary facilities, transportation infrastructure
Testing	Inspect/test to verify construction quality, safety, and compliance with regulation Potentially include pilot-scale demonstration prior to full-scale operation

Some milestones are conditional on success in others. For example, the establishment of responsibility and designation of a site are foundational to all other milestones. Facility design and the preparation of the safety analysis report and environmental report are important inputs to the licensing review.

However, not all milestones are tightly connected. Although planning for transportation should occur early, as it may influence siting, the establishment of transportation infrastructure can be considered somewhat separately from facility design and licensing because materials will likely not require transportation until many years after a license is granted to start construction.

The “cross product” of potentially required facilities (see Figure 1) and high-level milestones relevant to the deployment of each facility creates a conceptual “matrix” of facilities and milestones, diagrammed in Figure 2. This conceptual matrix can be applied to the direct-disposal back-end system for any reactor by considering the unique characteristics of its SNF, the management operations it requires, and where/how that management is expected to be challenging. In general, management “challenge” is considered qualitatively as perceived difficulty relative to the management of LWR SNF. Through this process, consideration of SNF management challenges is taken beyond a purely design and technical perspective and considered in terms of factors that influence the deployment of the management facilities themselves [5].

Milestone / Facility	Establishing	Siting	Transportation	Facility	Licensing	Construction	Testing
On-site storage							
Fuel Treatment							
Off-site storage							
Transportation							
Disposal							

Considering advanced reactor SNF management requirements, which combinations of facilities and milestones pose challenges?

FIG. 2. Diagram of conceptual “matrix” combining generic advanced reactor SNF management operations/ facilities with high-level milestones for facility deployment.

3. FRAMEWORK APPLICATION TO ADVANCED REACTOR SNF MANAGEMENT

This section presents high-level discussions about SNF characteristics, management requirements, and observed milestone challenges for five different reactor types. Four potential SMR system technology types are considered with regard to the SNF management path presented in Figure 1: sodium-cooled fast reactors, high-temperature gas-cooled reactors; liquid-fuel molten salt reactors; and lead-cooled fast reactors. Then, microreactor SNF management is discussed, considering alternative management pathways.

3.1. Sodium-cooled fast reactors

Sodium cooled-fast reactors (SFRs) use metal fuels, typically U-Zr or U-Pu-Zr, with stainless steel cladding. Fuel pins are arranged in hexagonal bundles and inserted into stainless steel ducts. A cross-section diagram of an example SFR fuel assembly is shown in Figure 3. The fuel may use sodium as a thermal bond to improve heat transfer between the fuel pin and cladding. SFR fuel designs that omit the sodium have been proposed: the fuel is manufactured to contact the cladding before irradiation and the fuel pellets are annular to accommodate swelling [7]. Because they operate at higher temperatures, SFRs achieve higher thermal efficiencies than LWRs. SFR SNF may also reach higher burnup than LWR SNF. As such, discharged SNF mass and volume per electricity generated may be lower for SFRs than for LWRs. However, SNF decay heat and radiation dose rate may be higher [5].

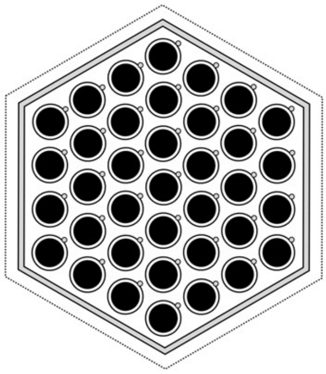

FIG. 3. Cross-cut diagram of example SFR fuel assembly.

SFR SNF will require on-site storage after discharge, final disposal, and transportation capabilities. Off-site interim storage is also a possibility. The U.S. has decades of experience storing and transporting sodium bonded fuel from various government-operated research reactors.

SFR SNF may pose disposal challenges. If the SNF uses sodium bond, treatment to remove it may be required prior to disposal because sodium is chemically reactive with water. Various treatment procedures are available [5]. The milestone associated with establishing responsibility for the treatment facility is a crucial concern. Whether or not fuel treatment to remove sodium bond is necessary in all cases is not clear; if direct disposal of sodium bonded SNF is pursued, further research and development will be necessary. Metal SNF may experience faster dissolution in groundwater than oxide SNF [8], which could impact repository performance.

All SFR SNF management facilities will need to accommodate its unique features, including elevated decay heat, radiation dose rate, and potentially lower structural integrity due to radiation damage. Sodium-bonded SFR SNF may require special handling.

Finally, there is little-to-no experience in licensing the types of facilities or operations required for managing commercial SFR SNF. Facilities or operations for which licensing would be new or expected to be notably more challenging for SFR SNF than for LWR SNF include on-site storage, fuel treatment (if necessary), transportation, and off-site interim storage. If fuel treatment is required and its waste products are classified as high-level waste (HLW), the current NRC regulations in 10 CFR Part 72 [9] would likely prohibit storage at a consolidated interim storage facility (CISF) if licensed as an independent spent fuel storage installation (ISFSI).

3.2. High-temperature gas-cooled reactors

High-temperature gas-cooled reactors (HTGRs) will likely use fuel forms such as graphite prismatic blocks and pebbles, both based on TRI-structural-ISOtropic (TRISO) coated fuel particles. Examples are diagrammed in Figure 4. HTGRs operate at higher temperatures and higher thermal efficiencies than LWRs. Because TRISO particles are structurally and thermally robust, HTGR SNF can reach higher burnup than LWR SNF. Therefore, the discharged SNF mass can be lower than that from LWR. The fissile loading in HTGR fuels is relatively low, so the SNF volume discharged can be much higher. For that same reason, decay heat and dose rate are generally much lower [5].

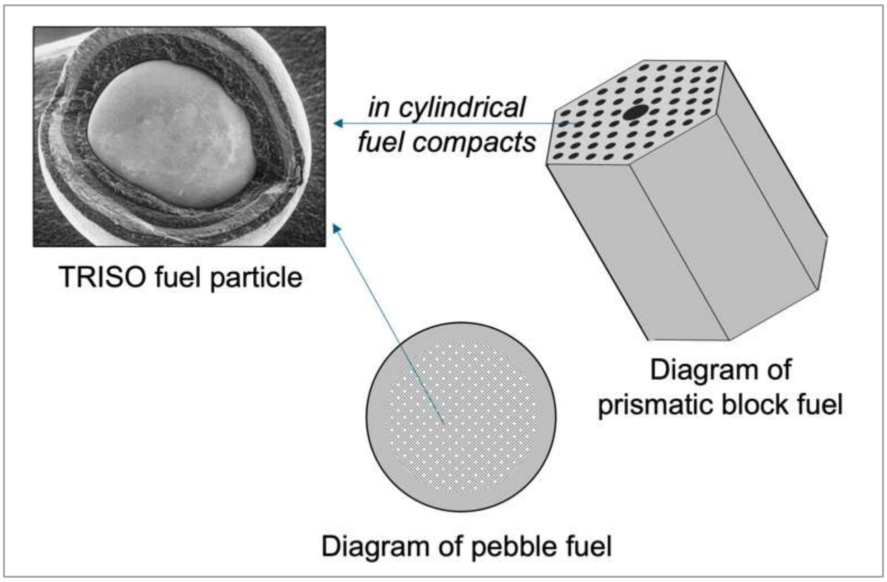

FIG. 4. TRISO fuel particle microscope image [1] and diagrams of HTGR fuel forms (not to scale).

HTGR SNF management will require on-site storage, disposal, and transportation capabilities. Off-site interim storage is also a possibility. Due to the relatively low heat density, high-temperature durability, effective heat transfer properties, and chemical and structural durability, management could be straightforward. Past research has evaluated TRISO-based fuels to be sufficiently robust for final disposal [11,12]. One potential challenge is for material accountancy and physical protection of pebble SNF during on-site storage at pebble-bed reactor sites. Pebbles are a unique diversion risk relative to LWR SNF due to their small size, regular movement, and lower radiation dose rate. This has implications for on-site storage facility design and licensing.

3.3. Liquid fuel molten salt reactors

Liquid-fuel molten salt reactors (MSRs) use molten salt as both coolant and fuel. MSRs can vary in terms of their salt chemistry (e.g., fluoride vs. chloride salt), fuel type (Pu-, enriched-U-, or Th/U-233-fueled), and neutron spectrum (thermal vs. fast). MSRs operate at high temperatures and can achieve higher thermal efficiency than LWRs. Because the fuel is liquid and cladding radiation damage is not a concern, some MSRs can achieve higher burnup than LWRs, but this is reactor dependent. Therefore, MSRs can discharge more or less the same amount of SNF heavy metal mass than LWRs [5]. The relative discharged volume is unclear and depends on the final waste form. Volatile and noble fission products will be removed from the fuel salt during operation and therefore will not be present in the spent fuel salt, affecting decay heat and dose rate.

MSR SNF management will require on-site storage, disposal, and transportation capabilities, will likely require fuel treatment, and off-site interim storage is also a possibility. Past experience storing spent fuel salt at the Molten Salt Reactor Experiment (MSRE) has shown the need to plan for salt instability: radiolysis drives the production of halogen gases that can form volatile compounds with actinides, allowing them to migrate through the system and leading to contamination [13]. This affects facility design for on-site storage.

Fuel treatment is likely to be required because salt is water-soluble and may not be acceptable for disposal in a geologic repository. The public or private organizations responsible for financing and executing fuel treatment facility deployment should be identified. There are many promising options for MSR SNF treatment [14], but significant research and development is necessary. MSR wastes will have unique characteristics that need to be accounted for in all management facilities.

Due to the lack of experience with commercial MSR technologies and SNF forms, licensing of new facilities and operations is anticipated to be challenging. Operations for which licensing is new or expected to be notably more challenging for MSR SNF than for LWR SNF include on-site storage, fuel treatment, and transportation. MSR waste forms, if classified as HLW, may be prohibited by NRC regulations in 10 CFR Part 72 [9] from storage at ISFSIs. If the CISF is licensed as an ISFSI, this could preclude acceptance of MSR waste forms.

3.4. Lead-cooled fast reactors

Lead cooled-fast reactors (LFRs) use oxide (UO_2 or U-Pu MOX) or nitride (UN) fuels. Like SFR fuels, fuel pins are hexagonally arranged and bundled into ducts. The pins may be backfilled with helium (as indicated in the analysis presented in Ref. [15]) or use lead as a thermal bond (as used in Ref. [16]). Because they operate at higher temperatures and thermal efficiencies and can achieve higher burnup, LFRs may discharge lower SNF mass and volume per electricity generated than LWRs, but decay heat and dose rate could be higher [5].

LFR SNF management will likely require on-site storage, off-site interim storage, disposal, and transportation capabilities. Design and licensing for all facilities will need to account for elevated LFR SNF decay heat and dose rate relative to LWR SNF. Fuel treatment is not anticipated to be required, but there are potential factors that may necessitate it. First, if lead is used as a fuel-clad bond, it may need to be removed before disposal because its toxicity could cause RCRA concerns. Second, the stability of UN in water is unclear and past research has produced mixed results. Some experiments have observed UN degradation rates in water to be significantly higher than those for oxide fuels [17,18]. An additional factor to be considered is that if the nitrogen used in the UN fuel is not enriched in N-15 it may accumulate C-14, raising source term concerns. Therefore, the acceptability of UN SNF for disposal should be evaluated. These factors could also complicate licensing for on-site storage and final disposal.

3.5. Microreactors

Microreactors are only loosely related by their size, which is typically in the range of 1-20 MWth. Most concepts are advanced, non-LWR reactors. Common characteristics include long cycle lengths and the use of high-assay low-enriched uranium (HALEU) fuel. Most microreactors would be designed for power production at remote sites. Otherwise, all of their other features, including their SNF characteristics, can be different. This section considers microreactor SNF management as a whole and focuses on the impacts of size, portability, and remote siting. Specific SNF characteristics may still pose management challenges but those would likely also apply to larger versions of that reactor type.

A microreactor's size and unique deployment mission create new pathways for SNF management, diagrammed in Figure 5. After shutdown and a cooling period, the reactor, indicated by "RX" in the figure, and its SNF would be removed from the site. This could happen in one of two ways. One possible pathway is to ship the microreactor vessel (with fuel inside) to a centralized facility for defueling. Alternatively, fuel handling equipment could be deployed to the site to defuel the reactor, allowing for separate transportation of the reactor vessel and its SNF. These pathways then converge, sharing consolidated interim storage and disposal. Finally, a third option is that the microreactor vessel is never defueled and is ultimately disposed of along with its fuel. All three pathways have advantages and challenges, primarily affecting on-site storage and transportation.

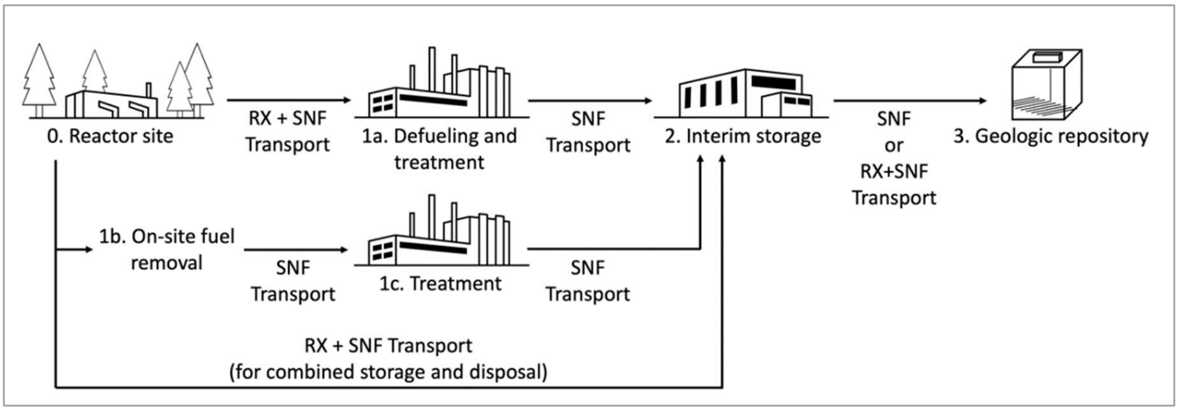

FIG. 5. Hypothetical options microreactor SNF management for direct disposal fuel cycle (reproduced from Ref. [5] with permission).

Combined transportation could allow for combined storage and disposal, but this could require new transportation, storage, and disposal systems to accommodate an entire reactor. A package capable of accommodating the size and weight of the fueled microreactor vessel would need to be designed and licensed. If sited in a remote area, there may not be transportation infrastructure (e.g., rail lines) available to transport such a large package. It is unclear whether the reactor would need to be shipped with its coolant to dissipate decay heat. Criticality safety is an important concern: the reactor would likely need to be shown as remaining subcritical if flooded with umbrated water. Altogether, these factors strongly affect the establishment of transportation infrastructure.

Separate transportation of the reactor and its SNF would greatly simplify transportation because existing transportation, storage, and disposal infrastructure can be more readily utilized. However, on-site defueling is a new operation that brings new challenges. Defueling equipment could be deployed with the microreactor or could be mobile and deployed at decommissioning. Defueling necessitates containment, may require dealing with reactive materials in the fuel or coolant (e.g., sodium), and raises questions about how to manage damaged fuels. All of this complicates microreactor deployment at remote sites, primarily affecting the design and licensing of on-site storage facilities.

Finally, nearly all types of microreactor fuel will have higher initial enrichments than conventional LWR fuels but will achieve lower burnup; as a result, greater quantities of fissile material will be present in the SNF at shutdown. Criticality safety, material accountancy, and physical protection are important design and licensing concerns and will affect every aspect of the SNF management pathway.

3.6. Overarching observations

The challenges for managing advanced reactor SNF are not equally distributed among all facilities or operations, nor among the milestones required to deploy them. Facilities for operations early in the management process – on-site storage and fuel treatment, specifically – have larger barriers to deployment because they needs to account immediately for the challenging characteristics of different SNF types. Facility siting, construction, and testing are important milestones but appear to be similar in difficulty for the management of any advanced reactor SNF type. Conversely, milestones such as establishment of responsibility, facility design, and licensing are anticipated to be challenging given the unique characteristics of each advanced reactor SNF design.

In addition to technical design work, addressing the management challenges for each of the advanced reactor SNF types could require one or more of the following: (1) NRC rulemaking in one or more parts of the relevant regulations; (2) Congressional legislation; and/or (3) unique physical security considerations.

4. CONCLUSIONS

This paper documents efforts to plan for the deployment of facilities that manage advanced reactor SNF in the direct-disposal fuel cycle. By considering the unique characteristics and anticipated management requirements of different types of advanced reactor SNF alongside high-level deployment milestones, qualitative insights into factors that influence the success of back-end system development can be made. This framework was applied to four different advanced reactor types – SFRs, HTGRs, MSRs, LFRs – as well as microreactors as a general class. Overarching observations about how advanced reactor SNF challenges affect back-end facility deployment milestones were noted.

This assessment is preliminary, high-level, and not exhaustive. It represents a first step in the planning process toward managing advanced reactor SNF. Importantly, none of the identified challenges should be viewed as disqualifying for any advanced reactor type. All can be accommodated by research, design, planning, and regulatory reform. However, these challenges should be considered early in the reactor development process so that none become burdens in the future.

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OPPORTUNITIES TO OPTIMIZE SMALL MODULAR REACTOR (SMR) USED NUCLEAR FUEL MANAGEMENT IN THE UNITED STATES

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Abstract

Innovation in used nuclear fuel storage technology has had a long and rich history in the US. Creative approaches became a necessity when the US cancelled its reprocessing programme in 1977. Today, the culture of innovation and the considerable industrial infrastructure it has created stands for the management of the various types of used fuels that are expected to be discharged from SMRs.

1. INTRODUCTION

The US has had a long and rich history of innovation in used nuclear fuel storage technology. Creative approaches became a necessity when the US cancelled its reprocessing programme in 1977. This meant that used nuclear fuel, which was originally thought to be destined to be shipped off-site for reprocessing, would need to remain at power reactor sites until a repository could be developed. The US did not initiate a repository programme until 1982, and then made only limited progress before terminating the programme in 2011. As of this writing, the date at which used fuel will be removed from US commercial power reactor sites remains unknown. Yet the use of nuclear energy in the US continues to flourish and is enjoying a resurgence in public support, spurred on largely by its reliability and carbon-free generation benefits. This is also due, in no small part, to how US industry has responded to the challenge of at reactor used fuel storage – by developing a suite of at reactor storage technologies that began as unanticipated solutions to an unexpected problem and is now seen as one of the most impressive industrial success stories of the past half century.

2. DISCUSSION

Today, the culture of innovation and the considerable industrial infrastructure it has created stands, ready to meet the new challenge of developing technologies for the management of the various types of used fuels that are expected to be discharged from SMRs. In understanding the future opportunity that this represents, it is important to first reflect on where we have been in the past. Throughout the 1980s and 1990s technological advances in storage rack design and the deployment of ever more sophisticated neutron absorber materials enabled spent fuel pools originally sized to store less than two cores of fuel to be able to store more than 10 cores [1]. As the ability to expand spent fuel pool storage capacity approached physical limits, reactor operators began, starting in 1986, to deploy dry cask storage systems that enabled fuel to be removed from the pools for placement on land adjacent to the plants in Independent Spent Fuel Storage Installations (ISFSIs).

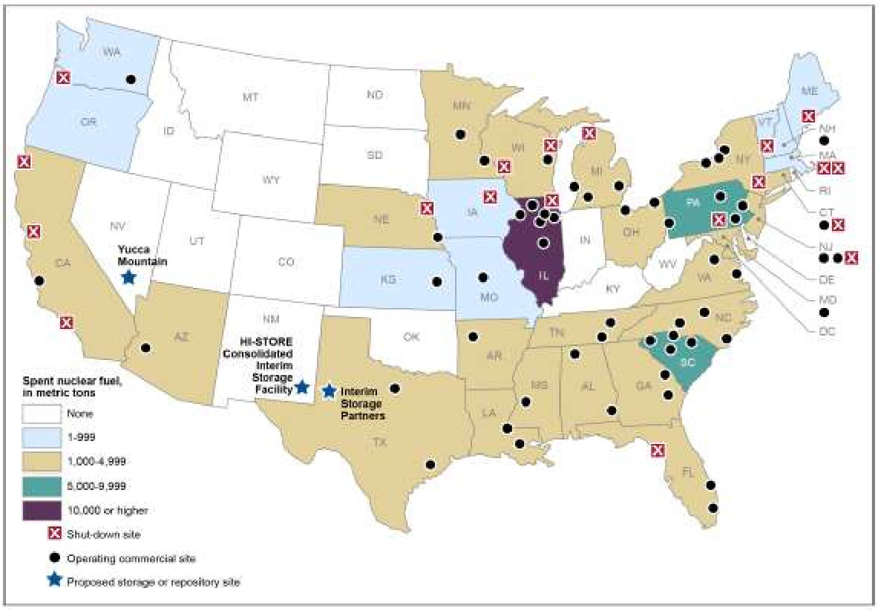

FIG. 1. Stored Commercial Spent Nuclear Fuel Amounts, through 2019, and Locations, as of June 2021[2].

As of July 1, 2022, 3,767 dry storage systems had been deployed at 74 ISFSIs in the US [2] (Figure 1). And with the move from wet to dry storage, the pace of technology innovation only accelerated. Initially licensed for 20-year periods, the licenses for more than a third of these systems have been extended to 60 years with many more 40-year renewals pending [2]. These extensions have been supported by the deployment of state-of-the-art aging management programmes [3] which are likely to further extend the service life of these systems in the future. In addition to life-extension, dry storage design innovations have been deployed to increase system capacity by more than 50% (despite the physical size of the systems being limited to what will fit on a rail car for transportability) and accommodate advances in fuel design and higher burnup fuels [2].

This culture of innovation has been driven by the highly competitive nature of the US used nuclear fuel storage marketplace. Currently three suppliers – Holtec, Orano, and NAC International – are well positioned to meet US storage needs and continue to offer new and improved storage solutions in the quest to gain market share. They each possess highly sophisticated and proven analytical tools for evaluating every aspect of storage system performance (i.e., criticality, heat transfer, shielding, materials issues, etc.) that they can continue to refine and apply to the design of transportation and storage systems for SMR used fuels. And the competitive pressure driving innovation is likely to become more intense in the future as storage cost considerations become a bigger factor in SMR owner decision-making. Operators of the current US reactor fleet get reimbursed by the government for most of their used fuel storage costs because, in accordance with the Nuclear Waste Policy Act of 1982 [4], they all signed contracts with the US Department of Energy (DOE) in 1983 for the Department to begin removing used fuel in 1998 for disposal. Since this did not happen, the reactor owners are able to recover storage costs through litigation and settlements. However, the used fuel removal contracts that the DOE will be offering to SMR owners will be very different. Since the 1998 date is no longer relevant, DOE has amended its standard contract to specify a performance date of "10 years after the original term of the plant's operating license or any license extensions [5]. This means that SMR owners, unlike the owners of the current fleet, will have to cover all used nuclear fuel storage costs for the life of the plant.

Other than the rather significant difference in how it is funded, the used nuclear fuel management marketplace for SMRs initially might not look much different that the marketplace of today. Three US light water reactor SMR suppliers (Holtec, Nuscale, and GEH) are all planning to use low enriched uranium similar to what is in the cores of the current fleet [6]. For these cases, storage innovation will simply be a matter of accommodating new configurations and perhaps higher burnups, although Nuscale has indicated that average burnup for their 250 MW thermal VOYGR design will be 45 GWd/t with a design basis maximum of 62 GWd/t which is very much in line with what is typically observed in the existing fleet [7].

Many of the designs not using light water as coolant will be using High-Assay-Low-Enriched-Uranium (HALEU) fuel [6]. Spent fuel discharges from these plants should be met with a storage innovation curve already in progress as a number of the reactors in the existing fleet are also planning to use variants of HALEU (often referred to LEU+) and will already be ordering storage systems before they will be needed by the SMRs. Fast reactors, such as Oklo, are mostly planning to use metal fuels for which there are already well-established transportation and storage technologies in use for discharges from DOE's metal fuelled research reactors. These technologies have been applied to the higher burnup higher enrichment fuels that may be used in some SMRs.

High Temperature Gas Reactors are planning to use TRISO fuels that, while characterized by higher enrichment and burnups, also provide a particularly robust multi-barrier containment including a stable silicon carbide layer that will last for more than a million years and graphite that does not degrade even in water [8] that should be amenable to a wide range of storage (as well as disposal) approaches. Molten salt reactors pose a unique set of materials related storage challenges. However, a significant amount of work has already been devoted to waste form and salt treatment options and scientists have expressed confidence that technical solutions are available [9]. Moltex's plans to deploy molten salt reactors in the US and Canada include a comprehensive fuel and waste strategy [10]. The use of sodium coolant in some SMRs also presents materials challenges due to sodium bonding with the fuel, a topic that DOE has already devoted significant effort to address [11]. It will be interesting to see the answers that a competitive marketplace will be able to bring to bear in response to these challenges.

One thing that has the potential to significantly change the used nuclear fuel storage marketplace for both the current fleet and future SMR owners used fuel is managed is, ironically, a reversal of the very decision that triggered the US innovation journey described above in the first place. Reprocessing (now more commonly referred to as recycling) is returning to the US used fuel management portfolio. In August 2020, a survey of 13 SMR suppliers found that at least 6 are interested in using recycled used fuel from existing US power reactors as feedstock for their machines while at least 8 are interested in recycling their own spent fuel [12]. Specifically, Oklo intends to submit a license application for a recycling facility to the US Nuclear Regulatory Commission in 2025 [13]. The re-emergence of recycling as part of the US fuel cycle will create opportunities for the production of tailored waste forms designed with long-term storage, transportation, and disposal in mind. In this sense, it could even make the innovators task easier.

Of course, prospects for the development of a repository for final disposal in the US remain far off. The national conversation on how to proceed is just now beginning anew with DOE recently having solicited and received public input on a Request for Information on using a consent-based siting approach to identify sites for interim storage of used nuclear fuel [14]. And while it is disappointing that the US has not been able to make progress on the repository development plan laid out in 1982, it is important to point out that at that time the US was a nation that was ramping down its commitment to nuclear energy. A number of new builds were being cancelled in the face of an economic downturn, and public support for nuclear energy had dropped following the 1979 accident at Three Mile Island. The repository design (Yucca Mountain) that resulted from that plan was based on a very specific population of used nuclear fuel and high-level radioactive waste. Now that the US is again ramping up its commitment to nuclear energy, with the deployment of a new generation of advanced reactors (SMRs) the disposal question is taking on a different shape. SMRs deployed along with recycling technologies have the potential to redefine the very nature of used nuclear fuel, both for that discharged from SMRs as well as much of the existing inventory. As the US again returns to answering the disposal question, it will be doing so with the benefit of over 40 years of highly competitive innovation in used fuel management technology in hand. The demonstrated success of the innovators should provide high confidence that future disposal solutions will be sound.

If one looks far enough into the future, it is not hard to see the frustration of present-day engineers and scientists who have laboured for decades to develop a repository for final disposal of used fuel, only to see that goal move farther away, replaced by the admiration of future historians who look back across centuries with gratitude that our civilization took the time to get it right. It is also not hard to imagine that SMRs will be the catalyst for that change in perspective.

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Paper ID#41

ROMANIA'S STRATEGY FOR RADIOACTIVE WASTE AND SPENT NUCLEAR FUEL IN THE CONTEXT ACCORDING WITH THE DEVELOPMENT OF THE EXPENDING NUCLEAR PROGRAMME

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Abstract

Romania is one of the 14 European Member States that retain the option of using nuclear energy. Currently, electricity generated by nuclear fission accounts for about 18% of the country's electricity production through two Cernavodă units. In Romania, spent fuel and radioactive waste is produced by civilian nuclear power generation, research, industrial, and medical activities. Romania has no military or defence programme to produce spent fuel and LILW-LL. The government's policy is to repatriate all spent fuel from the research reactors under agreement. There is no deep geological repository currently operating in Romania, so existing management arrangements for SNF and RW management focus on pre-disposal waste treatment and interim storage, and on preparatory works to implement geological disposal. An updating national strategy for safe management of SNF and RW in Romania will need to take into consideration an economically feasible solutions for integrating CANDU fuel cycles with new generation plants as well as strategic options on how to address the alternative of spent nuclear fuel recycling or conditions for the implementation of DUPIC cycle in Romania, taking into account the most recent national and international experiences.

1. INTRODUCTION

Nuclear energy is a sustainable alternative for the development of the national energy sector, given the limited resources of energy raw materials and the need for electricity generation that does not produce greenhouse gas emissions. The role of nuclear energy is critical to achieving the ambitious targets for reducing greenhouse gas emissions, while ensuring an effective contribution to energy security, an affordable price to end consumers and, an improvement in the country's energy independence.

The draft energy strategy for the period 2019-2030, which takes account of energy needs to 2050 [1], sets concrete targets, establishes clear directions, and defines the benchmarks by which Romania will maintain its position as an energy producer in the region and as an active and important player in managing regional stress situations. Strategic investment projects of national interest have been defined to enable the required development of the energy sector.

Planned Increases in nuclear energy capacities, include refurbishment of Unit 1 and completion of the design of Units 3 and 4 at the Cernavodă NPP as well as preparation of the infrastructure related to the development of small modular reactors, SMR type. Under this strategy, Romania will remain on of the 14 EU Member States that have maintain the option of using nuclear power. At present, the electricity produced from nuclear fission contributed about 18% of the country's electricity production. Following completion of the two new units at Cernavodă; the percentage rise to about 28% in 2035. As a low-carbon energy source, nuclear energy has a significant share in the total national electricity production and is a key component of the Romanian energy mix [1].

Nuclear energy in Romania is supported by domestic resources and infrastructure covering the entire open nuclear fuel cycle, giving Romania a high degree of independence in nuclear energy production. Analyses of the need to meet environmental and energy security objectives and targets, security of supply and diversification of sources for a balanced energy mix, ensuring the transition to a low greenhouse gas emission energy sector and an affordable energy price for consumers, demonstrated that the Cernavodă NPP Units 3 and 4 project is one of the optimal solutions to adress the electricity generation capacity deficit forecast for 2028-2035 when several existing fossil fuel-based units reach the end of their operational life.

2. NATIONAL STRATEGY FOR SAFE MANAGEMENT OF RW AND SNF IN ROMANIA

2.1. Current sources for spent fuel and radioactive waste

In Romania, spent fuel and radioactive waste are produced by nuclear power generation, and by research, industrial and medical activities.

Romania has one nuclear power plant, Cernavoda NPP, initially approved for five PHWR CANDU-6 type reactors with a 705 MW(e) gross capacity each, in different implementation stages. Unit 1 and 2 have been in commercial operation since December 1996 and respectively November 2007. The annual electricity generation by Cernavoda NPP Units 1 and 2 represents approximately 18 % of the overall electricity production of Romania, leading to the discharge of approximately 9975 tHM of spent fuel during 50 years of operation. Units 3 and 4 are under preservation, since 1992 and the construction of Unit 5 was definitively abandoned [2]. The CANDU reactors are expected to undergo refurbishment half way through their design life (after around 25 years of operation) during which some reactor internal components will be replaced [3].

At the end of 2020, Romania and the United States of America initialized an extended intergovernmental agreement for cooperation in relation to strategic projects for Romania such as Cernavoda NPP Units 3 and 4, the refurbishment of Unit 1 as well as in other civil nuclear projects. The intergovernmental agreement was ratified by the romanian parliament, in July 2021.

There are two nuclear research sites in Romania, both with research reactors and other facilities that generate radioactive waste:

— the VVR-S type research reactor located at the IFIN-HH research site at Magurele Bucharest has been decommissioned and regulatory authority issued the delicensing certificate in 2020. All HEU and LEU spent fuel from this reactor has been repatriated under agreement to Russian Federation.

— the TRIGA reactor located on the RATEN ICN site (Mioveni-Pitesti) has been converted to be operated with LEU fuel. All of the HEU fuel has been shipped back to the United States of America under agreement. Future spent LEU fuel generated from continued operation of this reactor is intended to be returned to its country of origin; as alternative, since currently no agreement is in force, its geological disposal in Romania is also considered.

“Both the IFIN-HH and RATEN-ICN research sites produce LILW-SL and LILW-LL from their own research activities and from facility decommissioning. They also collect radioactive waste and spent sources from other research, industrial and medical institutions around Romania” [3]. RATEN ICN also stores small amounts of spent fuel fragments from the spent fuel transferred from Cernavoda NPP for post-irradiation examination, as well as experimental fuel elements (both CANDU and TRIGA type) irradiated in TRIGA reactors. (1)

2.2. Potential future sources for spent fuel and radioactive waste

The construction of three more CANDU units (Cernavoda 3, 4 & 5) was started in the early 1980s but was discontinued in 1990. These partially constructed units were subsequently included in a conservation programme, in order to avoid deterioration and preserve the option for later use. Currently, it is estimated that the civil works, both in the nuclear and the conventional part of the facilities, are approximately 40% completed for Cernavoda Unit 3, 30% completed for Cernavoda Unit 4. Neither Unit 3, nor Unit 4 has any major equipment installed [1].

The government intends to complete Cernavoda NPP Units 3 and 4. For Unit 5 it was decided to change the use to accommodate emergency management and storage facilities. Construction and operation of new units, would increase the total volumes of generated. This report is based on estimate for spent fuel and radioactive waste arising from all 4 CANDU units [3].

SNN and NuScale Power have signed a Cooperation Agreement ("Teaming Agreement") - roadmap for the implementation of NuScale Small Modular Reactors (SMRs) in Romania, with the aim of developing a preliminary plan and roadmap for the development of a 462MWe NPP consisting of six 77 Mwe modules, with the possibility of expanding the number of modules and/or deploying modules on other sites).

Romania already undertakes research associated with Gen IV reactors. It also contributes to the development of the LFR technology by hosting and building the ALFRED demonstration reactor project and the experimental infrastructure supporting the testing, demonstration, qualification, validation, and verification activities required to authorize the demonstrator [2].

2.3. The system for safe management and disposal of spent nuclear fuel and radioactive waste

Under romanian law, the holder of a license to operate a nuclear facility is primarily responsible for the predisposal management of spent nuclear fuel and radioactive waste, as well as decommissioning and dismantling of the facility. The government has established the Nuclear Agency for Radioactive Waste (ANDR) to be responsible for disposal of spent nuclear fuel and nuclear waste.

The National Strategy includes construction of a new engineered surface repository for LILW-SL (the DFDSMA). This new repository is intended for the final disposal of the LILW-SL generated from operations, refurbishment and decommissioning of the 4 CANDU reactors at Cernavoda.

The agreed management end-state for spent fuel and LILW-LL is deep geological disposal, and a single repository will be constructed for all spent fuel arising from the 4 CANDU reactor units at the Cernavoda site, plus all LILW-LL [3].

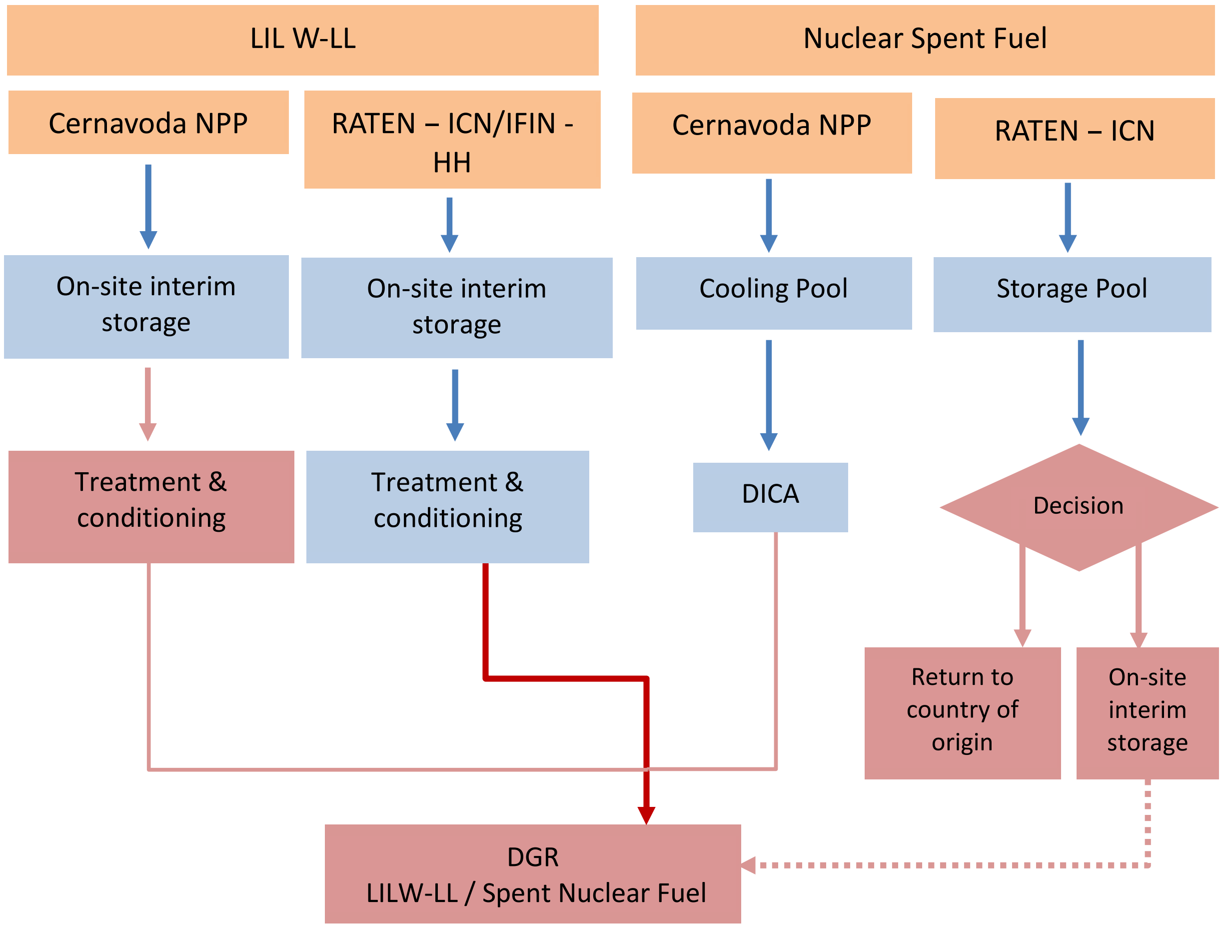

FIG. 1. The simplified management strategy for spent fuel and LILW-LL showing existing facilities and transfers (blue) and those that are planned for the future, subject to further approvals (red) [1].

2.4. Safe management of the long-lived low and intermediate level waste and spent nuclear fuel

The origin of this waste is primarily from research, industry and medical applications as well as from operation, refurbishment and decommissioning of Cernavoda NPP. The total anticipated volume of waste considered for geological disposal is currently estimated at about 42500 m3 (packaged volume) of which around 95 % is spent fuel from the Cernavoda NPP. Romania does not have military or defence programmes that produce spent fuel or LILW-LL. The government policy is that all spent fuel from the research reactors is to be repatriated under agreement.

There is no deep geological repository currently operating in the country, so existing management arrangements focus on pre-disposal waste treatment and interim storage, and on preparatory works to implement geological disposal. According to the national Strategy, operation of the Deep Geological Repository (DGR) is planned to start in 2055. The current spent fuel management strategy at Cernavoda site is based on interim storage. Spent fuel from the operating Units 1 & 2 is first held in cooling ponds (one at each reactor unit) for a minimum of 7 years, after which it is transferred to an on-site Interim Dry Storage Facility (DICA). The DICA facility is based on MACSTOR 200/400 type modules which are passively air-cooled. Currently, 13 MACSTOR modules have been licensed at the DICA facility, each with a design life of 50 years. It is planned to expand the facility in stages to store all spent fuel from both the existing and planned new-build reactors. It is government policy that geological disposal of spent fuel should be implemented as soon as is reasonably practicable, taking into account economic and societal factors, so as not to impose an undue burden on future generations. No repository design concept has yet been chosen but, for preliminary planning and costing purposes, the Canadian repository for spent CANDU fuel is taken as a reference [3].

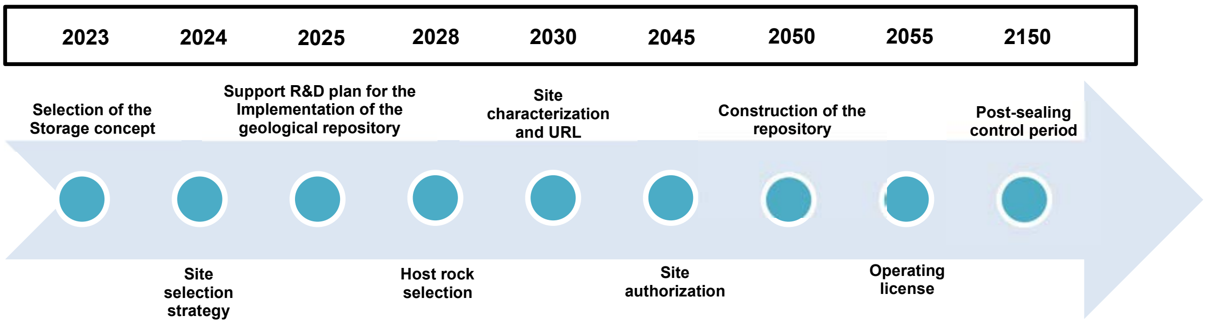

FIG. 2. The estimated implementation schedule of the geological DGR project.

The use of copper overpacks as in the Canadian reference design is expensive. A scoping study of alternative disposal concepts may identify less costly alternatives. In addition to the geological repository, a spent fuel encapsulation plant will also need to be built and operated. The design of the spent fuel packages will need to be consistent with the design of the overall repository concept (and site and host rock). There are substantial uncertainties in the costs for geological disposal, and these relate in part to the design and engineering concept. The Canadian concept assumed as reference design is based on copper canisters. The unit cost per canister is estimated to be €240,000 which means the total costs for the 3000 canisters needed would be €720 million. Considering the large inventory, disposal in a clay formation in steel canisters may be economically advantageous [4]. The location for the encapsulation plant has not been chosen, but it is reasonable to assume it may be constructed near to the DICA facility to minimize the effort needed to transport spent fuel.

Romania has over 25 years of experience in the safe operation of one of the world's most efficient power plants and a team of professional engineers, a world-renowned local engineering school and an extensive supply chain. The development of SMR power capacity based on zero-carbon technologies in place of a decommissioning coal-fired power plant at the earliest in 2028-2030 will contribute to Romania's efforts to phase out coal-fired power generation by 2032.

NuScale and Nuclearelectrica are taking steps to deploy the first NuScalse six module power plant (462Mwe) in Romania, this decade. In this context, SNN is carrying out a number of preparatory activities for the development of a first nuclear power plant based on SMR technology, including activities to assess the siting and suitability of various SMR supply chains. SNN is also involved in international multilateral activities to promote innovative nuclear SMR technology.

3. CONCLUSION

There are some risks and consequences associated with a management strategy that assumes 'early' disposal of spent fuel. Geological disposal is a significant undertaking and may divert attention from implementation of the DFDSMA which has more immediate programme implications for the Romanian nuclear sector. Geological disposal represents the single largest and most expensive component of any national radioactive waste management strategy. Planning for early disposal may mean there are insufficient financial resources or that resources need to be diverted from other activities. Geological disposal plans are extremely sensitive to external political, financial and stakeholder influences.

Taking into account the need to update the national strategy for safe management of SNF and RW, a number of activities are needed in the near future to find solutions for integrating the fuel cycles of CANDU and new generation plants. In this respect a series of analyses and studies are planned to inform future decisions on the safe management of spent nuclear fuel. These will consider integration of fuel cycles used in existing and future NPPs, spent nuclear fuel recycling, implementing a DUPIC (Direct Use of Spent PWR Fuel in CANDU) cycle in Romania and using spent fuel from CANDU reactors in the ALFRED project [2].

An update national strategy for safe management of SNF and RW in Romania will need to take into consideration an economically feasible solution for integrating CANDU fuel cycles with new generation plants as well as alternative strategies for spent nuclear fuel recycling implementation of DUPIC cycle in Romania. These studies will take into account of national and international R&D results for advanced CANDU fuel use.

For the time being, the strategy in Romania is the open fuel cycle without reprocessing. Both options have advantages and disadvantages. Given the new projects that are planned for the next decade, the management of the SNF will to be updated, drawing on output from R&D, studies and international experience.

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OVERVIEW OF THE UK'S TRANSPORT REQUIREMENTS FOR SMR BACK END FUEL CYCLES

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Abstract

Globally, over 65 Small Modular Reactor (SMR) designs exist at various stages of development; each of these designs implements a broad range of fuel cycles. In particular, the back-end consideration of these fuel cycles, including the packaging and transportation of the waste, needs consideration within the UK's existing nuclear infrastructure and capability. Nuclear Transport Solutions (NTS) has been involved in the back-end transportation lifecycle, from design, criticality and shielding analysis, to license security and operational considerations. This paper will provide an overview of the transport requirements for the SMR back-end fuel cycle within the UK, in line with associated international regulatory frameworks. Consideration will be presented in respect to transport safety, security, and operational requirements at a high level, and seek to highlight any factors that could be applicable to other member states.

1. INTRODUCTION

1.1. Background

Small Modular Reactors (SMRs) are defined as a nuclear reactor with a power output between 10 and 300 Mwe [1]. These reactors are typically small to medium in size and have been developed to utilize modular design and manufacturing principles, allowing for easier transportation and assembly. The various types of SMR designs encompass Generation III+ and IV technology, with the most mature SMR concept designs based on Light Water Reactors (LWRs). Other concepts include molten salt cooled reactors (such as molten salt reactors) and gas cooled reactors (such as High Temperature Gas-cooled Reactors (HTGRs)). Furthermore, the various SMR concept designs could utilize thermal or fast neutron spectrums, and could be land or marine (i.e., floating on water) based.

A variety of fuel types could be used across the range of SMR designs. Some will benefit from existing fuel concepts i.e., similar to fuels used in LWRs; whilst others are proposing new fuel designs such as TRi-structural ISOtropic particle (TRISO), High Assay Low Enriched Uranium (HALEU) (noting the Uranium-235 enrichment could be anywhere from 5 to 19.75 w/o) or molten salts.

Transport connects the points of the nuclear fuel cycle, with a case-by-case approach typically required depending on the fissile material in transport. The wide range of fuel cycles being proposed for SMR designs mean that some work will be required to establish the readiness of transport capability to support the deployment goals of SMR technology.

1.2. Nuclear Transport Solutions

Nuclear Transport Solutions (NTS) is a wholly owned subsidiary of the United Kingdom's (UK) Nuclear Decommissioning Authority (NDA), acting as the transport division of this group with shipping, rail and technical solutions capability. NTS is comprised of International Nuclear Solutions (INS), Pacific Nuclear Transport Limited (PNTL) and Direct Rail Services (DRS). PNTL are the world's most experienced nuclear shipping company, with 3 purpose-built vessels that have vast international experience transporting a range of nuclear material from across the fuel cycle (including MOX fuel, HEU, vitrified high level waste and spent fuel among others). DRS manage a fleet of over 100 locomotives that operate on the UK rail network undertaking nuclear operations frequently. In particular, NTS has vast experience transporting spent fuel by rail from the currently operating Advanced Gas-cooled Reactor (AGR) fleet Amongst other transport experience across the nuclear. The solutions capability within NTS provides the full lifecycle of package design and licensing, with extensive experience in new and re-purposed packages used to transport nuclear material.

2. UK OVERVIEW

2.1. Government Position

In 2020, the UK government set out a strategic case for new nuclear, which was detailed in the "Ten Point Plan for a Green Industrial Revolution" [2] and the Energy White Paper: Powering our net zero future [3]. This plan detailed the required increase in nuclear capacity with a view to accelerating the deployment of nuclear power within the UK. Great British Nuclear is being formed as part of these acceleration activities, with a Future Nuclear Enabling Fund (FNEF) also created offering up to £120m of investment in nuclear enabling efforts by the industry. The FNEF is the first in a series of interventions by UK government, designed to achieve the ambition of deploying 24GWe (three times current capacity) of nuclear power within the UK by 2050.

With the ambition to include more nuclear capacity using a variety of large scale, small modular and advanced reactor types, transport remains a crucial element of enabling the nuclear fuel cycle.

2.2. Rolls Royce Led Consortium - UK SMR

Currently only one SMR is going through the Office for Nuclear Regulation's (ONR) Generic Design Assessment (GDA), the Rolls Royce UK SMR [4]. The first UK SMR is expected to be connected to the grid in the early 2030s. The design is focused on optimizing the levelized cost of electricity against low capital costs, a common philosophy for modular designs.

TABLE. 1. ROLLS ROYCE UK SMR DATA [5]

Power Output (MWe)	470
Fuel Type	UO_2 pellet
Fuel Enrichment	4.95% (max)
Refuelling Cycle (months)	18-24

The UK SMR is a 470 MWe (above the IAEA generic SMR definition of 300 MWe) modular reactor, based on a PWR design, utilizing fuel of UO_2 fuel enriched to 4.95%. Spent fuel from the reactor will be transferred to an onsite spent fuel pool, after which the fissile material will be prepared for long term dry cask storage. The design decisions around the UK SMR appear to be focused on capitalizing on many of the technological and infrastructure assets already in place within the UK. The fuel cycle is expected to be similar to that utilized at Sizewell B (a PWR design). Due to these similarities, the back-end fuel cycle transport considerations for this reactor type are more a question of capacity (subject to the widespread deployment) as opposed to any technological needs or best practice and/or regulatory developments.

When estimating the capacity within the UK, the UK government has stated the intention to approve 8 new reactors by 2030. A broad estimate would see 3-5 of these being the UK SMR design. In addition, Rolls Royce are aiming to have 16 manufactured with various export opportunities being explored.

Initial siting indications are that the UK will use previously licensed sites to deploy these reactors at the outset, again capitalizing on what is already in place to suit a shortened deployment timescale. However, for multiple deployments it is not unlikely that new licensed nuclear sites could be seen within the UK. Road and rail transports are most common for the current reactor fleet, with the Advanced Gas Cooled reactor fleet spent fuel all being transported by rail.

3. TRANSPORT APPROACH

3.1. Regulation

(a) Safety

The goal of safety in transport of nuclear material and Other Radioactive Material (ORM) is the protection of people, the environment and property. The principal regulations for radioactive transport are the IAEA Regulations for the Safe Transport of Radioactive Material (SSR-6) [6], which form part of the IAEA Safety Standards series and supports the implementation of binding international instruments. These regulations are aimed at ensuring safety primarily by the package, irrespective of the transport mode. SSR-6 sits at the base of the hierarchy for the international and national regulations for the safe transport of radioactive materials, however SSR-6 is not mandatory. Above SSR-6, the UN Model Regulations cover all 9 classes of dangerous goods written in the form of model regulations, adopting the requirements of SSR-6 when concerning the transport of radioactive material (Class 7). The UN Modal Regulations are the mandatory, mode specific regulations that states are required to follow. These include:

— Road – The Agreement concerning International Carriage of Dangerous Goods by Road (ADR);
— Sea – The International Maritime Dangerous Goods (IMDG) code;
— Rail – The International Carriage of Dangerous Goods by Rail (RID);
— Inland Waterway – The International Carriage of Dangerous Goods by Inland Navigation (AND) (only applicable within the European Union);
— Air – The Convention on International Civil Aviation.

National or regional regulation concerning the transport of nuclear material or ORM is achieved through adoption of the Model Regulations or directly from the adoption of SSR-6. The regulations for the typical transport case for SMRs (utilizing licensed packages for moves associated with the fuel cycle) are well established and still relevant for the deployment of SMRs given the materials in use but will need close consideration when applied operationally for fuel types with less substantiation data.

(b) Security

The Convention on the Physical Protection of Nuclear Material (CPPNM) and its amendment [7] are the only legally binding international instruments in the area of physical protection of nuclear material. The original convention was signed in 1980, and more recently an amendment came into force in 2016. Whilst the original convention concerned the international transport of nuclear material, the recent amendment extended its application to nuclear materials in domestic use, storage and transport. The convention places several legal obligations on states who are party to the convention, to protect nuclear material in transport. Annex I of the Convention and its Amendment detail the levels of protection to be afforded to nuclear material based upon its Category. While Annex II (see Figure 1.) gives tabulated categorization of nuclear material based upon type, form and quantity. A key principle of nuclear security is that responsibility for nuclear security within a state rest entirely with the State. In the context of this paper, the Convention and its Amendment are still pertinent. Within the UK, the ONR operates an outcome focused approach to security regulation.

TABLE 2. ANNEX II OF THE AMENDMENT TO THE CONVENTION ON THE PHYSICAL PROTECTION OF NUCLEAR MATERIAL [7], DETAILING CATEGORIES OF NUCLEAR MATERIAL

Material	Form	Category I	Category II	Category III
1. Plutonium	Unirradiated	2 kg or more	Less than 2 kg but more than 500g	500 g or less but more than 15 g
2. Uranium-235	Unirradiated			
	- Uranium enriched to 20% U-235 or more	5 kg or more	Less than 5 kg but more than 1 kg	1 kg or less but more than 15 g
	- Uranium enriched to 10% U-235 but less than 20% U-235	n.a.	10 kg or more	Less than 10 kg but more than 1 kg
	- Uranium enriched above natural but less than 10% U-235	n.a.	n.a.	10 kg or more
3. Uranium-233	Unirradiated	2 kg or more	Less than 2 kg but more than 500 g	500 g or less but more than 15 g
4. Irradiated fuel (The categorization of irradiated fuel in this table is based on international transport considerations. The State may assign a different category for domestic use, storage, and transport, taking all relevant factors into account)			Depleted or natural uranium, thorium or low enriched fuel (less than 10% fissile content)	

3.2. Transport Safety Concept

Whilst there are various SMR designs and several proposed fuels, there will still be a requirement to transport fresh unirradiated fuel to the reactors and spent SMR fuel to a long-term storage solution, i.e., the Geological Disposal Facility (GDF) in the UK. The transportation of such spent fuel will require the selection of a suitable transport package, design of the package, including any shielding considerations, criticality assessment of the package(s), security considerations and infrastructure (covering the proposed mode(s) of transport, such as, road, rail, sea, or air).

To ensure the safe transport of this fissile material, it will need to be contained within a package suitable for road, sea, rail, or air (or could encompass a combination of these modes of travel) transit and demonstrate safety under the IAEA SSR-6 Regulations [6]. These regulations currently set out conditions that the package(s) needs to demonstrate under various routine, normal and accident conditions. A package licensing team will initially assess the proposed contents of the SMR spent fuel and assess the hazard categorization of the material. Depending on the hazard categorization, the team can use the SSR-6 Regulations to determine the most suitable package type to transport the fissile material. Once the package has been determined, the assessment of a package is undertaken in the Package Design Safety Report (PDSR), with contribution from various disciplines such as licensing, design and analysis, shielding and criticality, see Figure 1.

The licensing team will oversee the production of the PDSR and provide additional duties such as:

— Confirm the regulatory requirements needed;
— Confirm the contents of the proposed package;
— Implement Learning from Experience (LFE) from previous packages;

— Provide a crucial communication link between the PDSR authors and the competent authority (CA) i.e., the Office for Nuclear Regulation in the UK.

Typically, it is important to allow for early engagement with the competent authority (CA), especially if there are novel or complicated approaches needed with the licensing of the package. Further, ensuring all parties in the supply chain are Suitably Qualified and Experienced Personnel (SQEP) to be undertaking their role in supporting the production of the PDSR is also important.

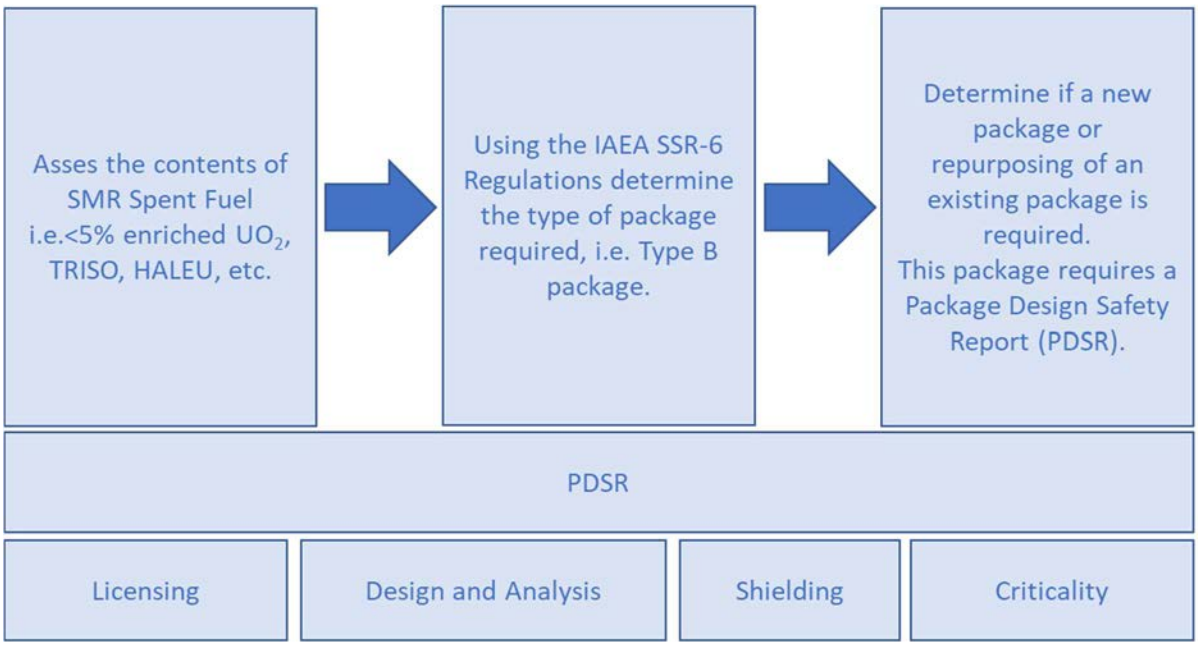

FIG. 1. Summary of the process to license a package to transport back-end SMR fuel

The Design and Analysis team perform the role of designing the package and take into consideration any limits imposed due to impact, structural, thermal, stress, shielding and criticality analysis. In particular for the Design and Analysis of the package there are no huge differences between the transportation of other fissile material and SMR spent fuel. To allow for the design of a new package or begin efforts to repurpose a package, this team will require information such as:

— The proposed mass of material per package;
— What the material(s) are;
— The size of the material to be transported ;
— Whether there is potential for interface of the package between various plants.

Analysis teams (such as thermal, impact, shielding and criticality), as per the IAEA SSR-6 Regulations [6] needs to assess assessment of routine, normal and accident conditions of transport. In particular for the criticality and shielding assessments, a large amount of data needs to be collated to allow for the production of suitable models, such as:

— Design drawings;
— Fissile material contents;
— Fuel details;
— Normal and Accident Conditions of Transport (NCT and ACT) damage conditions;
— Stress conditions;
— Temperature profile;
— Fuel break up;
— Fissile release.

The shielding and criticality assessments will branch from here, with the shielding assessment ensuring limits set out in IAEA SSR-6 Regulations [6] are adhered to.

In particular for the criticality assessment, the initial assessment covers NCT, which covers minor accidents, or mishandlings other package during transit i.e., a package falling off the back of a wagon or locomotive. The NCT criticality assessment will investigate:

- Tolerances to parameters such as dimensions of the fuel pellets, fuel pins, assembly and package etc;
- Material variations such as mass, enrichment and density of the fissile material and any moderator;
- Results of NCT damage conditions (obtained from the thermal, stress and impact analysis), such as drop orientations, relocation of fissile material etc;
- In addition, the factors of criticality needs to be assessed on top of any NCT variations (mass, absorbers, geometry, interaction, concentration, moderator, enrichment, reflection, volume).

For ACT Assessment, which conditions under which the package sustains damage that is equivalent to that from a severe and credible accident, the criticality assessment will further consider conditions which are a result of typical regulatory tests or design and analysis tests covering:

- Fully engulfing fire (up to 800°C);
- 9 meter drop onto unyielding target;
- 1 meter drop onto a punch bar;
- Immersion in water.

These tests could result in multiple end states – all of which require assessment which could result in the consideration of:

- Rearrangement of the contents;
- Reduction of spaces within or between packages;
- Loss of efficiency of neutron absorbers;
- Water leaking into or out of the package;
- Water immersion;
- Temperature changes.

In addition to these potential end states, the factor of criticality needs to be assessed on top of any ACT variations (mass, absorbers, geometry, interaction, concentration, moderator, enrichment, reflection, volume)

4. ADVANCED FUEL TYPES

In lieu of specific data regarding the composition of spent fuel in the advanced fuel types listed below, broad assumptions have been made accounting for the fuel characteristics that have been stated in publicly available sources. To date, no design utilizing the following advanced fuel types has been announced within the UK, however, it is expected that they will be deployed in country in the future. Table 2 demonstrates that a number of SMR designs are proposing the use of these novel fuel types.

TABLE 3. LAND BASED SMR FUEL DATA [5]

Total Fuel Type Entries	66
Enrichment listed > 10%	35
MOX or Plutonium	8
TRISO	19
Molten Sat	5

4.1. TRISO

TRISO fuel is comprised of an inner sphere of fuel and a ceramic outer sphere, which has specifically been designed to exploit enhanced safety characteristics, such as accident tolerance. TRISO fuel is a good example of a fuel type that is less dispersible and considered too large and dense to travel in the atmosphere except under extreme conditions, which may prove beneficial in transport safety and security planning.

4.2. Molten Salts

Molten salt fuel types are the concept of dissolving fissile or fertile material within a salt for use in a molten salt reactor. This also creates interesting and currently unanswered questions for transport requirements. Further information specific to this fuel type is going to be required to determine the appropriate transport requirements that will satisfy both safety and security.

Both fuel types are seeing significant interest due to, among other qualities, enhanced safety characteristics within a reactor. These safety benefits are also expected to be present for transport, while simultaneously posing benefits for transport security. Due to the expected inherent safety of these fuel types, a different approach may be adopted for transport when assessing the requirements within the SSR-6 Regulations [6] and of the CPPNM and its Amendment [7]. As before, further substantiation is required for both fuel types to satisfy the transport planning requirements.

5. GENERAL CONSIDERATIONS

5.1. Challenges Associated with the Transport of SMR Spent Fuel

The most prominent barrier to the roll out of SMR deployment is the licensing process. Currently the regulations are designed towards ‘standard’ conventional reactors i.e., there are clear definitions of the material and packages being transported. For SMRs, there needs to be clear and agreed definitions of what SMRs are, and a clear scope of the material being transported. Due to the ongoing developments around various reactor designs, the specific data mentioned throughout Section 3.2 is not currently available. As designs near deployment, the ability to conduct more specific analysis to determine capability and asset requirements is expected to be established, however there is much work to be done in the area.

Most of the SMR designs are still in the conceptual phase, with fuels such as TRISO and HALEU having less experimental data and substantiation than the more widely used UO_2 fuels in LWRs. Similarly, this will mean that efforts to substantiate the fuel’s safety characteristics against the transport requirements will need to be undertaken well before the deployment.

Package availability remains ambiguous until further work is conducted on the specific fuel cycle. When data regarding the material that can be expected in the back end of SMR design is available, the associated package requirements can be determined and mapped against the assets currently available within the UK. At this point, decisions regarding repurposing old packages vs designing new bespoke packages can be taken.

When considering transport security, the first step in forming the transport Physical Protection System (PPS) is to categorize the material based upon its typed form and quantity. This means that similar questions are present with regards to material characteristics substantiation. From the perspective of security, no immediate challenges appear present for SMR types using the already established fuel types (UO2), apart from the lack of data to estimate the security requirements to support an increased programme of spent fuel moves within the UK. Work is already ongoing to design the requirements associated with transports to the GDF.

For the advanced fuel types, it is useful to note that security is focused on two event types: theft and sabotage. Theft has the objective to gain control of the material at some point, whereas sabotage aims to produce radiological releases to the environment or public. A theft scenario will typically have two phases, the first being overcome the PPS and obtain the nuclear material, the second being to escape said material. Sabotage will typically have only one phase "defeating the protection of the nuclear material by means of weapons or intrusive tools and creating a radiological hazard." [8]. For theft scenarios, the specific recoverability of the fissile material within the fuel type will underpin the appropriate security arrangements required to be built into the PPS. The more easily the fissile material can be recovered from whatever form it is within the fuel or spent fuel, the more attractive this fuel will be to malicious actors for theft. Data to determine this in both the TRISO and salt fuel type could demonstrate a specific resistance to recoverability that has not been seen with previous fuel types in use (UO_2 or MOX etc.). If an additional chemical process is required to separate fissile material out of the fuel in transport, the fuel types could be considered inherently more security than predecessor fuels.

For sabotage scenarios the dispersibility (or release fraction) will be of most interest when measured against security events. These fuels contain the fission products within the ceramic of chloride / fluoride salt, therefore demonstrating an expected inherent resistance to sabotage scenarios, especially when compared to conventional fuels or any powdered fissile material. Therefore, some of the designed safety features are expected to benefit transport security, yet these are still to be further substantiated.

6. FUELLED TRANSPORTATION NUCLEAR POWER PLANTS (TNPPS)

The concept of moving nuclear material contained within a core module (either as fresh fuel or carrying spent fuel) is seeing much interest currently by the international community. The wide array of deployment potential is a key benefit of these smaller, more transportable reactor designs. Harnessing this deployment potential further through the concept of TNPPs presents profound benefits when considering areas of poor grid connectivity, disaster relief scenarios and even military application. Current transport regulation will need to be examined to determine the applicability of their regulation to this scenario, but at the outset some key considerations come to light for both transport safety and transport security.

6.1. Safety

Many questions exist regarding the *state* of the reactor core in transport while containing irradiated nuclear fuel. A series of assurances would be required to demonstrate the chance of a criticality occurring in transport is not possible. It is assumed this would be achieved most efficiently by securing control rods positions within the reactor core, however this would also need to be substantiated against the various requirements stipulated within SSR-6 [6], which are discussed within Section 3.2. The criticality parameters and assessments previously applied to a licensed package will have to be applied to the core, which would likely see work taking place on the core to place it into a transport compliant state. Following this there would be a need to ensure that all safety features remain in place within the core throughout the entire transport duration.

Sealed cores could see an operational lifetime of 25+ years, any safety assurance activities conducted at the start of the lifecycle could be redundant by the time the TNPP is required to conduct the return transport. This could be for mechanical / technical reasoning due to degradation of material over time or due to a change in the regulatory requirements in transport over the operational period of the reactor. In both instances, the level of safety assurances carried out at the front end of the life cycle of the reactor, will not necessarily lessen the assurances required at the back end of the reactor lifecycle. Therefore, establishing elements such as material performance characteristics of the module still meeting the transport SSR-6 Regulations [6] testing requirements after a prolonged period of operation will be necessary. If these modules / cores do not meet the requirements, decisions on what can be implemented in transport to ensure the transport requirements are met can be taken (such as increasing robustness with specialized overpacks etc.)

Dedicated analysis specific to each transport will be required, due to the range of variables and options (preferred mode of TNPP transport, refuelling cycle, module characteristics etc.)

Essentially, a consensus needs to be found on whether the particular factory fuelled reactor in transport should be considered an operable reactor or seen as a licensed package of nuclear fuel. This decision would depend on the assessment of the technical measures implemented in a particular TNPP design to prevent criticality in transport and satisfy transport safety requirements. Moreover, these decisions would need to be implemented and agreed by consignee and consignor state, plus any transit states entered during the transport.

Also noteworthy is the International Code for the Safe Carriage of Packaged Irradiated Nuclear Fuel, Plutonium and High-Level Radioactive Wastes (formerly the Irradiated Nuclear Fuel (INF) code) was renamed in 2021. This code specifies the activity limits associated with the rating of the vessel, with INF-3 being the highest rating bearing no activity limit. For maritime transport of loaded cores, vessel availability could be impacted by the contents of the core under current frameworks, when measured against the activity limits specified within the code.

6.2. Security

Categorization would still form the first step in developing the appropriate PPS for a TNPP. Assessing the attractiveness of the nuclear material to a malicious actor would influence the relevant security arrangements when considering the threat of theft. A key security consideration when returning a reactor module containing spent nuclear fuel is more likely a sabotage event. The fission products that will be present in the core present the highest likelihood of creating Unacceptable Radiological Consequences (URCs). Additionally, there may be residual heat and pressure within the core making the design more susceptible to a sabotage event. In any case, specific analysis of the core measured against the relevant Design Basis Threat (DBT), or threat assessments would be necessary to justify any security claims of the module. Given the vital area analysis that would be conducted for a licensed nuclear site, data from this analysis could also inform the anti-sabotage arrangements during transport for a loaded core module. Smaller TNPPs would lend themselves to being more easily secured during transport. This is due to the ability to overpack the reactor core or contain it within a form of high security iso-container purpose built for transport.

Considering that all security systems will perform one of, or multiple, of the functions in Figure 2, the idea of remote siting and remote transport in these cases will be underpinned by a fundamental principle. Ultimately the identified delay time (i.e., the delay a malicious actor of accessing material) needs to always be greater than that of the response time associated across the entire transport route. This principle may present opportunities to incorporate a secure by design approach to TNPP development to address this, or to utilize centralized response arrangements for multiple sites.

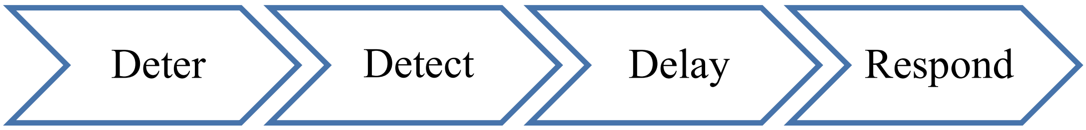

FIG. 2. Security system functions.

For a core module containing TRISO or molten salt fuel types, should these fuels demonstrate a continued ability to contain the fission products, the radiological consequences may be determined to be below what the state would consider a URC. In this case, less onerous security approaches to sabotage in transport may be justified.

7. SUMMARY

The wide range of fuel cycles proposed creates an issue for assessing general transport capability requirements. For SMRs utilizing the already established fuel types, SMR back-end fuel cycles should be able to capitalize on the infrastructure and assets already in place, however the lack of specific data on the material makes it impossible to confirm this. The increase in use of HALEU will need further work to substantiate it against the transport requirements, data availability (particularly for those enriched above 10%) is also not available. This would also identify the package requirements for the fuel type. Advanced fuel types are also in need of further substantiation for both transport safety and security requirements; however, it is noteworthy that the proposed safety benefits seen within operation will also be beneficial in transport. Finally, the concept of moving nuclear material within a reactor core is new and demonstrates promising innovation efforts with regards to harnessing the full deployment potential of some micro advanced reactor types. Further detailed investigation is required in this area to realize the concept, yet it could be a huge advantage when considering areas of poor grid connectivity, energy security and disaster relief scenarios. Efforts should be undertaken to conduct a gap analysis of this concept against the transport regulations, potentially identifying a new regulatory and requirements for moving nuclear and radioactive material within a core module.

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BACK END OF THE FUEL CYCLE CONSIDERATIONS FOR CAREM – THE ARGENTINE SMALL MODULAR REACTOR

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Abstract

CAREM-25 is a prototype reactor of 100 MWt and 32 MWe, and CAREM Commercial Units will have 120 MWe. It is a PWR type, which has passive safety systems, natural circulation, self-pressurized and low-enriched UO_2 fuel. The prototype is in Lima, Zárate, Buenos Aires Province, where Atucha's Nuclear Power Plants are established. Fuel elements are designed with a hexagonal section. There are 61 fuel elements in the reactor core, and its refuelling is annual. The spent fuel storage pool is inside the containment building and may store the spent fuel elements from 10 years of full power operation. Interim storage for radioactive waste will be provided at the CAREM site and the radiological characterization will be determined by direct, semi empirical, or analytical methods. The anticipated options for managing spent fuel from the CAREM reactor are similar to the others NPPs in Argentina: wet storage during the time necessary to allow sufficient decay of the fission products and later interim dry storage on the reactor site.

1. INTRODUCTION

The Argentine Atomic Energy Commission (CNEA) was created in 1950, initiating research and development activities in basic areas. In the following years, progress has been made with the development of nuclear technology, the operation of relevant facilities working on the production of radioisotopes for medical and industrial applications and the performance of tasks in connection with the nuclear fuel cycle, including mining and uranium processing activities, manufacturing of fuel elements for research and power reactors, production and generation of nuclear power, production of heavy water and the operation of three nuclear power plants. In addition, reprocessing programmes were undertaken at demonstrative scale.

Concerning the radioactive waste management, the legal framework "is stated by the National Constitution and legislation issued by the National Congress, mainly by Law No. 25279, which approved the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management; Law No. 25018, by which CNEA is appointed the implementing authority to perform all activities related to radioactive waste management and sets up the National Radioactive Waste Management Programme (PNGRR); and by Law No. 24804 which regulates Nuclear Activity" [1]. This law assigns CNEA the state ownership of spent fuel and makes CNEA responsible for the management of radioactive waste.

The current Nuclear Power Plants in Argentina are three: EMBALSE (PHWR - 648 MWe), operating since 1984, has undergone a life extension (2016 - 2018); ATUCHA I (PHWR - 357 MWe), in operation since 1974 (planned outage in 2024, for an estimated time of 2 years, what will allow to extend its operation for an additional twenty-four years at full power); ATUCHA II (PHWR - 745 MWe) in operation since 2016.

The projects under development are: CAREM 25 (MWe CNEA Design), under construction in Atucha site; ATUCHA III (HPR-1000 PHWR - 745 MWe) under negotiation with the PRC for the construction and regarding the technology transfer contract for the nuclear fuel assemblies to be built in Argentina.

There are also currently five research and radioisotope production reactors in operation (RA0, RA1, RA3, RA4 and RA6) and one under development (RA10).

As it has been stated in the Joint Convention Report, "in Argentina, spent fuel due to its fissile contents is considered to be a potential energy source and not radioactive waste. The National State is the owner of the special fissile materials contained in the spent fuel." [1]

The management of spent fuel comprises: firstly a wet storage in pools at the facilities to let the fission products decay and secondly interim dry storage. Spent Fuel at Embalse Nuclear Power Plant is stored in pools for six years minimum, before being transferred to dry storage.

In the case of Atucha Nuclear Power Plant Unit I, the capacity of the wet storage at the facility was provided until 2015, so in 2012 the ASECQ Project was initiated for dry storage of spent fuel. It has been completed in August 2022 and it comprises dry vertical silos for 2844 fuel elements.

For Atucha Nuclear Plant Unit II, spent fuel is stored in wet storage at the pools until completion of ASECQII, which is currently under conceptual and basic engineering design for dry storage.

Concerning the spent fuel from research and radioisotope production reactors, it is stored in pools at the reactor sites and in the case of reactor RA-3 it is later transferred to a new wet spent fuel storage facility at Ezeiza Atomic Center.

It needs to be noted that all the Spent Fuels from the Research and Power Reactors which contained Highly Enriched Uranium (HEU) provided by the United States have been returned.

Concerning the final disposition of spent fuel and high-level radioactive waste, research and development activities for a deep geological repository are being carried out.

TABLE 1: CURRENT SPENT FUEL MANAGEMENT FACILITIES [1]

Site	**Facility**
CNA I	I & II Pool Building
CNA II	Pool Building (UFA)
CNE	Storage Pool
	Storage Silos (ASECQ)
CNEA	Storage facility of RA-1 fuel elements (DECRA-1)1
	Central Storage Facility for research reactors SF (DCMFEI)1,2
	Research Reactors irradiated Fuel Storage Facility (FACIRI)

[1]Facility from the Ezeiza Radioactive Waste Management Area (AGE).[2] Since March 21, 2019, the date on which the last MTR item was transferred to the FACIRI, this facility did not store CGRI

2. CAREM-25

CAREM-25 is a prototype reactor of 100 MWt and 32 MWe, and CAREM Commercial Units will have 120 MWe. As a PWR type, it has passive safety systems, natural circulation, self-pressurized and enriched UO_2 fuel (3,1 and 1,8% enrichment). It is designed to bring new solutions based on an integrated light water reactor, it is an indirect cycle reactor, simple and highly safe. The prototype is in Lima, Zárate, Buenos Aires Province, where Atucha's Nuclear Power Plants are established.

Fuel elements are designed with a hexagonal section that has 108 fuel rods, 18 guide tubes for absorbing elements and one instrumentation tube, 127 in total. There are 61 fuel elements in the reactor core, and its refuelling is annual.

The spent fuel storage pool is inside the containment building and may store the spent fuel elements from 10 years of full power operation, and will include a cooling and clean-up system whose functions are: "removing decay heat dissipated by SFs stored in the SF pool as a safety measure; if required, it will enable decay heat removal of a whole core once the reactor has been extinguished for 60 hours; keeping the radiological, physical and chemical parameters of the water of the fuel Elements pool within an appropriate range; compensating water loss by evaporation." [1]. The CAREM civil work is currently in more than 60%.

As stated in the Join Convention report, "the design of the solid radioactive waste management system complies with the ALARA principle. It includes collection, segregation, characterization, conditioning, and interim storage processes of the RW arising from the operation and maintenance of CAREM-25. RW will be managed to ensure an acceptable level of radiological protection of workers and public, and the preservation of the environment. RW to be generated in normal conditions in CAREM-25 will be low or intermediate level RW. The Solid Waste Management System will include equipment to perform tasks such as pressing, drying and immobilization." [1]

The interim storage for radioactive waste will be provided at the CAREM site and the radiological characterization will be determined by direct, semi empirical, or analytical methods.

As mentioned, the anticipated options for spent fuel for the CAREM are similar to the others NPPs in Argentina: wet storage during the time necessary to allow sufficient decay of the fission products and later interim dry storage in the reactor site. In the specific aspect of the latter, in order to use the ASECQII under construction, it is still necessary to evaluate shielding and cooling for CAREM fuel due to its burnup degree and activity, compared to that of current PHWR, depending also on how long they stay in the decay pools. This is one of the aspects in which the current fuel cycle strategy might be modified.

The infrastructure needs for developing and implementing the foreseen strategy for managing CAREM spent fuel, considering CAREM spent fuel represent a very small amount compared to the other power reactors has not yet been analyzed.

Concerning research and development needed to support the development and implementation of the anticipated strategy for managing CAREM spent fuel, design of new shielding, transport systems and handling equipment could present challenges and issues to be taken into account, making use of the existence of developed tools for the treatment and handling of fuel from the CNA1 and CNA2.

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Paper ID#45

MANAGING FUEL FROM SMRS: UK FRAMEWORK AND HTGR GAP ANALYSIS

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Abstract

The UK government sees nuclear as playing an important role in delivering its strategy to achieve net zero by 2050 [1]. The current strategy includes investments in large nuclear stations, SMRs and continued support for fusion. This strategy is supported by substantial funding for R&D and skills development.

1. NATIONAL SMR PROGRAMMES

The UK hosts a number of SMR developers, and the government is supporting SMR development through a number of programmes. These include some funding for the UKSMR being developed by Rolls Royce and two rounds of funding for non LWR SMR concept development by a range of commercial organisations. Most recently, the government has identified HTGRs as a technology of particular interest for net zero transition and has initiated a programme to develop a demonstration reactor by the end of the decade.

In the UK, management of spent fuel is as commercial decision for the owners, subject to meeting regulatory requirements. For existing reactors and large LWRs under construction, fuel is being managed using an open cycle, employing a combination of wet and dry storage. Spent fuel reprocessing is however not precluded, therefore all fuel cycle options remain open for future SMRs.

The regulatory system in the UK includes a process for technical approval of reactor designs prior to site specific license application, known as Generic Design Assessment (GDA) [2]. This process involves nuclear (ONR) and environmental (EA) regulators and is staged to provide advice to applicants as the design matures. The process requires applicants to provide details of spent fuel management arrangements as part of the design documentation. These arrangements need to demonstrate a viable pathway to final disposition and the expected level of detail increases with design maturity. With respect to disposability of fuel and radioactive wastes, the regulators seek advice from the waste management organisation responsible for development of the UK's national repository (Waste Management Services) to assess the submissions using the process established for assessing the disposability of existing spent fuels and intermediate level wastes [3].

The disposability assessment process involves a comparison of packaging proposals for spent fuel or waste against package specifications that have been derived from the UK's Geological Disposal Facility (GDF) safety case and design. The process provides stakeholder confidence that materials can be packaged in a manner that is compliant with GDF design assumptions, a route to adapt GDF concept/design if required and it provides a mechanism to ensure that spent fuel management and disposability issues are integrated into licensing and permitting processes.

2. INITIAL HTGR GAP ANALYSIS

As part of the UK government's Advanced Fuel Cycle Programme (AFCP), NNL have undertaken a structured initial assessment of the current state of knowledge on prismatic HTGR behaviour relevant to its post-discharge management and disposition in order to identify potential priority areas for research to underpin the back end of a fuel cycle for HTGR deployment [4]. Consideration of pebble-bed HTGR was only omitted on the basis on insufficient time. Whilst differences are expected to be small, there is a desire to address this in the future.

The feasibility of reprocessing HTGR fuel was reviewed in a parallel activity. Whilst details are not captured here, the overall conclusion was that the technology was not sufficiently mature to enable reprocessing to be a reliable baseline option that could support initial reactor deployment.

The analysis approach was to consider each component of the prismatic fuel block in turn and identify expected degradation mechanisms relevant to the component. For each degradation mechanism, assessments were made of the adequacy of knowledge about the degradation mechanisms, considering particularly knowledge about the time frame for initiation of degradation, the rate of degradation under expected conditions (of storage, transport, or disposal) and the consequences of degradation on safety. Considering the knowledge of the consequences of degradation, potential impacts on key safety functions were evaluated and the steps or stages in the spent fuel management system, in which conditions could induce the degradation and the consequences could be significant, were identified.

A qualitative ranking method was used to identify the priority areas for further work by classifying areas where further data were required as High, Medium, or Low according to the technical needs to underpin spent fuel management for HTGR deployment. The output of the assessment is summarised below:

TABLE. 1. THE OUTPUT OF THE ASSESSMENT FOR FURTHER WORK [4]

	Fuel Kernel	**Containment Layers**	**Graphite matrix**
High Priority	-	PyC corrosion (aqueous) SiC corrosion (aqueous)	-
Medium Priority	Oxidation (gaseous) Corrosion (aqueous) Fission gas release Helium release	Fission product migration Helium Pressurisation PyC oxidation (gaseous) SiC oxidation (gaseous)	Oxidation (gaseous)
Low Priority	-	Fission Product Attack Propagation of existing flaws	Corrosion (aqueous)

The prioritised items in the contents of the table are dependent on some of the characteristics of the selected bank-end spent fuel management option. For this assessment, it was assumed that the fuel blocks would not be dismantled and that an optimised disposal concept for HTGR was considered to be attractive. The reasons for this are discussed below.

3. FUEL BLOCK DISMANTLING

Prismatic HTGR fuel blocks are mostly composed of graphite rather than fuel. Removal of the fuel compacts, or the fuel particles themselves, would greatly reduce the volume of material requiring disposal as fuel. Whether this is attractive or not depends on many factors, including packaging and disposal requirements for the separated graphite, whether the concentration of heat generating materials in spent fuel packages results in packages with characteristics that affect pre-disposal storage requirements and emplacement requirements, and whether the burdens introduced by additional processing have a significant overall impact. The impacts of relevance would include a range of factors including cost, environmental and operational safety and the balance will depend on the size and timing of expected arisings.

For this study, no dismantling was assumed, in the context of the expectation that irradiated graphite would be disposed of as ILW in a deep repository (the reference case for gas cooled reactor core graphite) and that the low volumetric heat generating of intact blocks would enable emplacement of fuel at short cooling times and with smaller spacing intervals.

4. DISPOSAL CONCEPT

The current disposal concepts for spent power reactor fuel are based on ceramic fuels LWRs that have also been shown to be suitable for steel clad Advanced Gas-cooled Reactor fuel. However, TRISO fuel containment layers have a high integrity and durability. If their performance in disposal can be qualified, then it may be possible to devise a simpler disposal concept for TRISO fuel that would reduce disposal costs whilst maintaining adequate safety and environmental protection. To assess whether such an approach is viable, exploratory work would be prioritized much more highly, as it is more urgent, than it would if the reference case is used because that case assumes that no containment is provided for the fuel matrix.

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TRISO FUEL MANAGEMENT DEPENDING ON THE CHOICE OF THE FUEL CYCLE – RESEARCH CURRENTLY CONDUCTED AT INCT IN POLAND

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Abstract

Gas-cooled high temperature reactors (HTGRs) are nuclear reactors that could reach high outlet temperatures (750 - 950 °C). They are Gen IV reactors for the future market, with efficient power generation and high temperature process. The fuel used in HTGRs are TRISO (TRI-structural ISOtropic) spherical particles. TRISO fuel is distinguished by its high resistance on mechanical and chemical actions like impacts, crushing, corrosive substances etc. These features make it appear safe and suitable for direct disposal but difficult to reprocess for actinide recycling.

In Poland the commissioning of the first HTGR reactor is planned within the next 10 years. The start-up of a new reactor needs a number of issues related to the safety of its operation to be resolved. Among them is safe management of the spent fuel once it has been discharged from the reactor. The Institute of Nuclear Chemistry and Technology (INCT) was involved in development of procedures for the HTGR waste and spent fuel management within the scope of the project "Preparation of legal, organizational and technical instruments for the HTGR implementation". Currently, the INCT in a consortium formed with partners from other European countries is implementing the Horizon Europe GEMINI 4.0 - GEMINI For Zero Emission project which aims to demonstrate high temperature nuclear cogeneration with the use of High Temperature Gas-cooled Reactors (HTGR). As part of this project, the INCT will review different options of closed fuel cycle with HTGRs: spent TRISO fuel reprocessing and recycling alternatives. First results of the studies are presented in the paper.

1. INTRODUCTION

In addition to the implementation of the Polish Nuclear Power Programme [1] aimed at building in the country,6 to 9GWe of installed nuclear capacity based on the pressurized water nuclear reactors of generation III/III+. In addition to the obligation to reduce carbon dioxide emissions, an important factor enhancing the development of nuclear energy in many countries around the world is the environmental aspect [2]. Launching the HTGR technology for supporting Polish industry and achieving climate goals is considered [3]. The fuel used in the HTGR reactor is TRISO, a fuel ensuring inherent safety of the reactor, but having a different structure from the fuels used today. The start-up of a new reactor needs to be preceded by complex analysis and solving a number of problems related to safety of its operation. The new generation of reactors impose new challenges for radiation chemists due to their new conditions of operation and the usage of new types of fuel and coolant/heat carrier [4]. One of the issues that should be resolved is, how to safely manage this novel fuel once it has been discharged from the reactor.

The HTGRs are the Gen IV reactors that may be applied in many industrial processes, e.g., in the production of various chemicals, oil refining, iron and steel metallurgy and hydrogen production [5]. They are characterized by efficient production of energy and supply the technological heat at high temperatures. The technology is considered as " high advanced" in terms of safety, economy, and environmental impact. The reactors are designed to withstand sudden increases in internal temperature in the case of a cooling system failure. The properties of the moderator (graphite) and the coolant (helium) do not change at high temperature and high radiation activity. Damage of the TRISO fuel does not occur even in the case of accidents such as multiple losses of reactor shutdown functions [6]. This is related to the structure of fuel elements. Each spherical element contains about 8300 particles that can be about the size of a poppy seed. A spherical fissile kernel (e.g., UO_2 or ThO2) is covered by the concentric pyrocarbon (PyC), and a silicon carbide or zirconium carbide (SiC) layers (Figure 1). The main advantage of the HTGRs is deep fuel burnout, as well as burning of the long-lived actinides and plutonium in the course of their operation, which results in reducing the toxicity and the amount of generated waste.

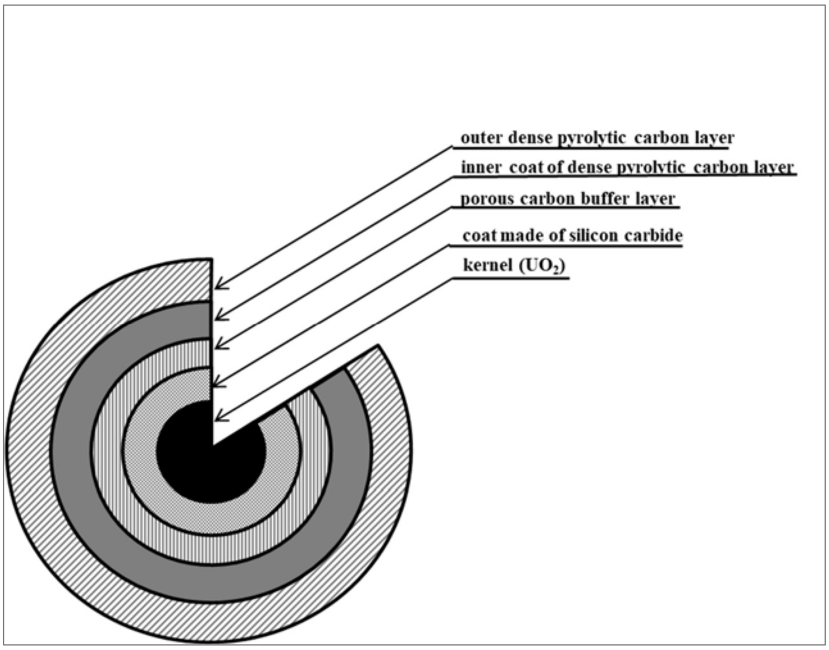

FIG. 1. The layers of TRISO coated fuel particle.

There are several methods for fabricating fuel kernels. The sol-gel process is the most commonly used process today [7]. At the INCT a new variant of a sol-gel method called—Complex Sol-Gel Process (CSGP) has been elaborated to obtain uranium dioxides. This method has been used to synthesise uranium dioxide (also doped by neodymium, as surrogates of trivalent plutonium and americium). Spherical particles (diameter below 100 μm) obtained by CSGP are presented in Fig. 2 [8].

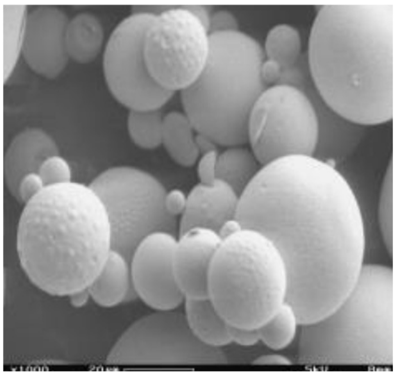

FIG.2. SEM analysis of uranium dioxide in the shape of spherical particles (diameter below 100 μm) obtained by CSGP.

2. OPTIONS IN TRISO SPENT FUEL MANAGEMENT

The essential assumption of the HTGR technology is to significantly reduce the production of nuclear waste through the appropriate design of reactors. However, some waste produced during the reactor operation is inevitable. During the HTGR operation, a flux of neutrons generates a number of reactions that lead to formation of the radioactive nuclides. When uranium fuel is used, most of the fissions come from the fission of U-253. Numerical modelling is used to simulate the generation of fission products with an increased burnup, and ORIGEN is utilized for depletion calculation of each fission product. Continuous fuel management is divided into five discrete periods for the feeding and discharging of fuel pebbles. Developed programs are used for discrete fuel management that traces 41 isotope nuclides, the production of which seem highly probable. In addition, the inventory of actinides at the end of each cycle is also investigated. The results will be informative for understanding waste management of spent fuel of HTGRs [9]. The core layers, and the fuel kernel, should be designed to decrease the impact of the burnt-up fuel on the environment. The potential release of these substances is stopped by the coating layers of TRISO fuel.

The methods of handling the spent nuclear fuel discharged from the reactor (back end of the fuel cycle) are related to the nuclear fuel cycle strategy being implemented. There are two main strategies: the open fuel cycle (once-through fuel cycle), when the spent fuel is not reprocessed, and the closed fuel cycle (twice-through cycle), when the spent fuel is reprocessed. In fact, for spent nuclear fuel three options for the management are considered. There are: reprocessing and recycling, direct disposal in a geological repository or the long-term, on-site storage until deep geological repositories are available.

The short-lived low level- and intermediate level waste (LLW and ILW, respectively) that are generated at all stages of the fuel cycle can be acceptable for disposal in the near surface facilities. The volume of waste should be reduced up to one third of the initial one by treatment or conditioning operations, like incineration, compaction, evaporation of the solvent, or another physical processing.

The spent nuclear fuel, in turn, in the beginning, is stored in the temporary storage facilities until the level of its activity allows the transfer to the final disposal site. The interim storage could be placed either in the vicinity of the nuclear reactor site, far away from the site (AFR), or at a centralized storage place [10]. In the centralized storage the monitoring is easier since many containers are placed at the same place. However, the transportation of spent fuel is strongly inconvenient. From the technological point of view, there are distinguished two kinds of interim storage facilities: wet or dry. Wherein, wet storage is the cooling pool next to reactor, so it is not usually the solution used at the AFR option.

2.1. Direct disposal of the TRISO spent fuel

Long-term storage of the spent HTGR fuel is a challenge to be met. The solutions that could be used to safely treat and dispose the spent TRISO fuel are still being studied. The tight TRISO coatings provide a potential highly resistant barrier that prevents the release of fission products and reduces the need for additional barriers. Moreover, this fuel is designed to reach very high burnup and to ensure a full retention of any gaseous fission product in every case of reactor operation (usual or anomalous, as well). It induces that direct disposal of the TRISO spent fuel can be easier than the management of the fuel from water reactors.

In the open fuel cycle, spent nuclear fuel is not recycled. It is discharged from the reactor, and then it is kept in a pool localized next to the reactor until the activity diminishes (usually, 5-10 years). After that, it is moved to the interim storage facility, where the storage is continued over 50-100 years. Finally, the spent nuclear fuel should be transported to the deep geological disposal site.

In the direct disposal, two ways of handling TRISO spent fuel should be considered. The first one, is the direct storage of entire graphite blocks. The main disadvantages of this solution are enormous volume of high-level waste that requires huge storage space and the need to be safely disposed for a long time. It is predicted that a single high temperature reactor will generate about 100 to about 170 tons of spent fuel per year.

An alternative option is separation of the particles of spent fuel from the graphite block. This conception has been developed in the United States and in France. The large blocks containing TRISO spent fuel have been extracted from the installations by using the mechanical methods. It is important to note, that the damage of the TRISO fuel coating was not observed. Then, in the case of the HTGR fuel particles, graphite-containing pellets are effectively separated using an electric field applied to the aqueous solution in the reactor. These TRISO particles are stable in this process. The separation of the TRISO particles from the graphite matrix will result in the significant reduction of volume of the high-level waste stored in a geological repository [11]. The immobilization of TRISO-UO2 particles in glass for an eventual geological disposal is also considered [12]. Low temperature vitrification of nuclear waste by sol – gel process has been developed at INCT [13].

2.2. The reprocessing of TRISO spend fuel

Reprocessing of spent fuel from the HTGRs provides meaningful reduction of the volume of high radioactive waste. The first step is to remove the graphite matrix from the fuel kernels. It allows for controlling the radioactive contamination of the graphite matrix. Then, the graphite matrix can be finally disposed as a medium or low-level waste. The separated compacts or TRISO particles can be conditioned for their direct storage or can be reprocessed for uranium and plutonium recovery.

Separation of layers of the TRISO fuel spheres from the fuel elements and decomposition of the spent TRISO particles was studied with using various, mostly mechanical techniques. The results were not satisfactory, even after removing all the graphite layers. High durability of the SiC coatings induced difficulties in release the uranium kernel [6]. The method proposed by Fütterer et al. seems to be promising, due to adaption of the high voltage head-end process to release of coated particles from the fuel element and to decompose the covered particles [14]. The tests were performed with crushed fuel pebbles containing unirradiated surrogate coated particles. In the first step, after one minute of exposure all coated particles were released. In the next step, the conditions were modified, and the coating layers and the remaining matrix material were decomposed to leave uranium kernel that can be directly subjected to chemical reprocessing. It is worthnoting that the energy consumption in this process is low.

Nowadays, the uranium price is relatively low, so it seems unnecessary to recover it from the spent TRISO fuel. Especially because the process seems more difficult than for spent fuel from light water reactors [3]. Moreover, it should be emphasized that the end-stage of the back end of the nuclear fuel cycle is the same in both mentioned above options (Figure 3).

Currently, the widely investigated option is a deep geological disposal. The facility is intended to be located under the ground, in the stable rock, which may to be assumed the natural barriers for isolation radionuclides from the biosphere.

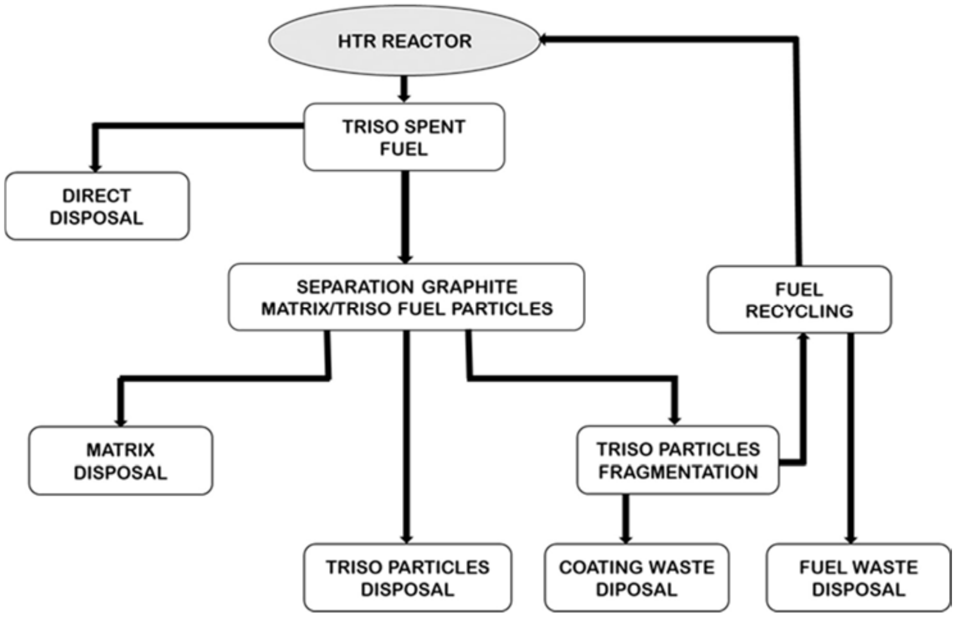

FIG.3. HTGR Fuel waste management pathways.

3. RADIOACTIVE WASTE MANAGEMENT IN POLAND – CURRENT SITUATION AND PLANNED STRATEGY

At present, there is only one MARIA research reactor in operation in Poland. It is the main source of the radioisotopes that are used in nuclear medicine, science, and many technological processes. The previous one, reactor EWA, operated for 37 years and was shut down in 1995. The spent fuel rods, discharge from the reactors, were temporarily stored in in the water pool of the MARIA reactor which is placed in the National Centre for Nuclear Research (NCBJ) and in the nuclear fuel (SNF) storage facilities operated by the Radioactive Waste Management Plant (ZUOP). The latter is the only Polish organization authorized to dispose the radioactive waste. Currently, there is only one radioactive waste repository in Poland that accepts the low- and intermediate level, short-lived waste. The long long-lived waste is accepted in the repository only for the temporary storage.

In December 2015, the Polish Council of Ministers accepted the National Plan of Radioactive Waste and Spent Fuel Management which was required by the European Commission. This Plan adopts into Polish law the Directive No 2011/70 of the Council of the European Union Community concerning the responsible and safe management of the spent fuel and radioactive waste. The crucial objectives of the National Plan of the Radioactive Waste and Spent Fuel Management are selection of the location, construction and starting operation of a new near surface radioactive waste repository (NSRWR) for the short-lived radioactive waste. In addition to that, the Plan mentions the need for work on the deep disposal for high-level waste.

It is necessary to note that the preliminary selection of the potential sites for the future deep repository for the high-level waste was made. Amongst them, there are five locations in the North-East part of Poland: clay deposit in Pogorzel and the magmatic formations in Krasnopol, Tajno, Kruszyniany and Rydzewo. Moreover, there were also found four other locations in the Central and Western Poland: three in the salt deposits in Damasławek, Kłodawa, Łanięta and one in the clay deposit - Jarocin (Figure 4).

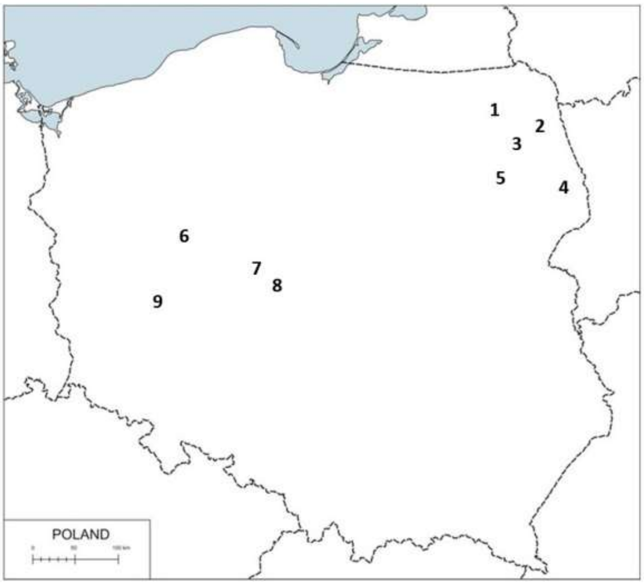

FIG. 4. The pre-selected locations of deep disposal of radioactive waste in Poland. 1- Pogorzel, 2- Krasnopol, 3- Tajno 4- Kruszyniany 5- Rydzewo, 6- Damasławek, 7- Kłodawa, 8- Łanięta, 9- Jarocin.

4. HTGR RESEARCH PROGRAMME IN POLAND

The document “Energy Policy of Poland until 2040 (EPP2040)” published in March 2021, includes the nuclear energy into the future energy mix [15]. According to the EPP 2040, HTGR is assumed to be a potential supplier of an industrial heat. The construction of the first industrial HTGR reactor a the capacity of 150-300 MWth is planned before 2031 [3,15].

At the beginning of this year, in the frame of the Strategic Programme financed by the National Centre for Research and Development (NCBiR): "Social and economic development of Poland in the conditions of globalizing markets" (GOSPOSTRATEG): "Preparation of legal, organizational and technical instruments for the implementation of HTGR reactors", a consortium of the Ministry of Climate and Environment, the National Centre for Nuclear Research and the Institute of Nuclear Chemistry and Technology has completed a new project [16]. Currently, the INCT in a consortium of 21 partners from European countries as well as from Japan and South Korea, is implementing the Horizon Europa GEMINI 4.0 - GEMINI For Zero Emission project which aims to demonstrate high temperature nuclear cogeneration with a High Temperature Gas-cooled Reactor (HTGR) [17].

The Gemini 4.0 project has been started in 2022 and is planned to run for 3 years. The Institute of Nuclear Chemistry and Technology will review different options for the closed fuel cycle with HTGRs: spent TRISO fuel reprocessing and the recycling alternatives.

At the current stage of the development of the Polish HTGR programme, an open fuel cycle with the temporary storage of the TRISO fuel and with no further processing is recommended. It is justified both by the technical and economic reasons.

5. CONCLUSION

Poland is developing the nuclear power programme with the purpose to start the operation of the first NPP around 2033.

At the same time, as parallel initiative, the HTGR technology will be developed. The implementation of two such significant programmes requires the development of a strategy for management of radioactive waste and spent nuclear fuel from nuclear power plants.

At present, the temporary on-site storage of spent nuclear fuel from Polish nuclear reactors is recommended. Then, it will be moved to the deep disposal facility (if it will be at that time available) or will be reprocessed. Numerous studies concentrate on the development of the procedures suitable for the HTR waste management. It seems that by improving the currently employed extraction methods of reprocessing the spent nuclear fuel, it will be possible to implement reprocessing in the 4th generation of reactors.

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FUEL CYCLE SCENARIOS AND BACK-END TECHNOLOGIES OF HTGR IN JAPAN

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Abstract

Japan has developed back-end technologies to establish a multi-recycling fuel cycle with fast breeder reactors (FBRs) to ensure energy resources. Even though the development of FBR has been focused on fundamental research, the reprocessing technologies for uranium fuel and disposal technologies had been completed for the Light Water Reactor (LWR) fuel cycle. These technologies were inherited to utilities and are about to be practical. Japan had been completed High Temperature Engineering Test Reactor (HTTR) a prototype and research reactor, a commercial High Temperature Gas-cooled Reactor (HTGR) design Gas Turbine High Temperature Reactor 300 (GTHTR300) with related reprocessing technologies and is planning a domestic demonstration reactor project. In this context, a representative fuel cycle policy is reprocessing in Japan. However, Japan has investigated various fuel cycle scenarios to expand the usage of the commercial HTGR. Then, we would like to introduce the scenarios and development status of related technologies in the present study.

1. INTRODUCTION

The objectives of the present study are to introduce the R&D of the back-end technologies with its history, fuel cycle scenarios of High Temperature Gas-cooled Reactor (HTGR) with related back-end technologies, and to specify the Research and Development (R&D) subjects, that contribute to the cooperation for R&D of back-end technologies of Small Modular Reactors (SMRs).

2. R&D HISTORY OF BACK-END TECHNOLOGIES

2.1. Technologies for LMR

Japan has developed back-end technologies to establish a multi-recycling fuel cycle with fast breeder reactors (FBRs) to ensure energy resources. Even though, the development of FBR has been focused on fundamental research [1], the reprocessing technologies for uranium fuel and disposal technologies had been completed for the Light Water Reactor (LWR) fuel cycle on the process.

The reprocessing technology had been developed by introducing Plutonium Uranium Redox EXtraction (PUREX) technologies from Cogema in France. Tokai Reprocessing Plant (TRP) [2] was built in 1977 as a research plant for LWR spent fuel reprocessing, and it was designed based on the UP2 design [2]. In addition, it is said that the part of the TPR design has been inherited by the UP3 design [2]. However, the process after the plutonium separation is different from the French design because Japan is a non-nuclear-weapon nation. The separated plutonium needs to be mixed into the same amount of separated uranium according to the Japan-U.S. reprocessing negotiation [3]. In 2006, an active test for commercial operation of Rokkasho Reprocessing Plant (RRP) was performed. RRP was designed based on the UP3 design.

The geological disposal technology had been developed by introducing the concept of KBS-3 [4] proposed by Svensk Kärnbränslehantering AB (SKB) in Sweden. The original concept is for direct disposal of LWR spent fuel, and the technology is applied to the disposal of vitrified waste. In 2014, a full-scale mock-up test of a deep geological repository started in Horonobe [5].

2.2. Technologies for HTGR

Japan completed High Temperature Engineering Test Reactor (HTTR) [6] a prototype and research reactor. A commercial HTGR design Gas Turbine High Temperature Reactor 300 (GTHTR300) [7] and related reprocessing technologies are planned to provide a domestic demonstration reactor project.

It is found from R&D history described in Section 2.1 that the representative policy in Japan is reprocessing. To adapt the reprocessing policy to HTGR, the head-end process of reprocessing for HTGR spent fuel had been designed and demonstrated [8]. The process is composed of five steps as follows: (1) extraction of fuel compact from fuel rod, (2) removal of graphite layer by burning, (3) graphite recovery from off-gas by CO decomposition, (4) removal of SiC layer by hard disk crusher, (5) recovery of tritium from a fuel kernel by voloxidation. The step (4) is dominant for recovery ratio, and the objective and target are crushing the SiC layer without crushing a fuel kernel. By adjusting the gap between disks of the crusher, 100% of SiC layer failure and 0% of kernel failure can be achieved.

The applicability and feasibility of RRP were confirmed, and the plant of head-end process is designed [9]. To apply to RRP, which is designed for LWR spent fuel reprocessing, the fuel composition is adjusted by diluted by 3.1 times with depleted uranium in order to match the composition to LWR spent fuel's one because the burn-up of spent fuel of HTGR is almost 3 times than that for LWR spent fuel. With these treatments, the spent fuel of HTGR can be reprocessed in RRP. We believe the technologies are already practical although a demonstration may be needed in the First-of-a-Kind (FOAK) plant.

Low Level radioactive Wastes (LLWs) from the reprocessing, which are categorized as TRansUranium (TRU) waste, is particular in HTGR reprocessing. The fuel blocks that are dominant contributor to the TRU waste amount, and that can be simply separated from the fuel element just by withdrawing fuel rods, because we selected a pin-in-block type fuel to reduce the reprocessing amount [8]. Fuel block disposal can be performed under current law in the existing LLW repository as shallow-ground pit disposal [10]. The criteria are set as the radioactivity concentration of waste for each radioactive nuclide. The radioactivity is generated by the activation of impurities.

Sometimes grafhite, impurities are purified by using chlorine gas or freons. However, residual Cl-35 is activated to Cl-36, and it is said that the radioactivity can become problematic [11]. In the design of GTHTR300, the fuel block is made of nuclear glade graphite IG-11 from the viewpoint of the economy because the IG-11 is not purified [12]. On the other hand, HTTR employs IG-110 for fuel block. In the future, it may be employed as the material of fuel block in GTHTR300 due to a design change. The IG-110 is also nuclear grade graphite and highly purified by halogen gas including chlorine, and the concentration of the chlorine remains as an impurity is lower than the detection limit of 3ppm [12]. Significant radioactivity cannot be generated from that concentration by activation.

3. FUEL CYCLE SCENARIOS FOR HTGR AND RELATED BACK-END TECHNOLOGIES

3.1. Fuel cycle scenarios for HTGR

As described in Section 2.2 the representative scenario for HTGR is reprocessing. As options, several other cases have been investigated. The first one is direct disposal with a uranium fuelled reactor [13]. This is more practical than the representative scenario of reprocessing. By virtue of the characteristics of the pin-in-block type of fuel, only fuel rods would be disposed of in a direct disposal canister. The canister number and its footprint can be reduced by 60% compared with an LWR case.

The second one is direct disposal with plutonium fuelled reactor [14]. This is a plutonium burner HTGR concept, so called Clean Burn, which is similar to Deep Burn [15]. Deep Burn can effectively incinerate the plutonium recovered from LWR spent fuel by fabricating oxide fuel without uranium. However, Deep Burn cannot be performed according to the Japan-U.S. reprocessing negotiation [3]. Then, we proposed mixing an inert matrix of Yttria-Stabilized Zirconia (YSZ) instead of uranium. Therefore, Clean Burn is limited only to direct disposal.

The last one is multi-recycle based on uranium [16]. This concept is aimed at toxicity reduction similar to FBRs and Accelerator Driven System (ADS) [17]. The HTGR design is in a range of thermal neutron reactors. Therefore, fissile uranium should be provided out of the fuel cycle. The actinide nuclides are recovered and recycled to fabricate fuel in the next cycle. The recovered uranium is also recycled and enriched.

3.2. R&D subject for HTGR to improve specification of fuel cycle

Basically, back-end technologies for the fuel cycle options had been completed, even if some of them should be demonstrated in a FOAK plant. However, to improve the specification of the scenarios, several subjects exist.

Commonly for all scenarios, the burden of graphite waste from the spent fuel should be reduced. To this end, there are two ways, one is a rationalization of near field model to evaluate public dose from the disposal repository. As described above, the waste can be disposed of by shallow-ground pit disposal. However, this evaluation is just the comparison of inventory of the radioactivity with the limitation which is determined by radioactive nuclides migration calculations under the condition of LWR waste. The source term of the waste is assumed to be dissolved into groundwater from the beginning. This condition is too conservative for the HTGR graphite waste due to its highly-water proof characteristics. If a proper near field model of HTGR graphite waste should be developed, the waste may be disposed of more easily from shallow-ground pit disposal to shallow-ground trench disposal without developing hardware.

Ultimately, the fuel block should be reused. To reuse the graphite block, irradiation damage needs to be recovered by annealing or refabrication from pulverizing process [18]. These technologies need to be demonstrated.

To realise the multi-recycle based on uranium, the uranium and plutonium are recovered by reprocessing and Minor Actinoids (MAs) are recovered from the High Level Liquid Waste (HLLW) by separation. Recovery ratios also depend on the specification of the head-end process described in Section. 2.2. Moreover, to satisfy the criteria of toxicity reduction, by which the toxicity decays to natural uranium level in 300 years, a recovery ratio of more than 99.9% is necessary [19]. Therefore, the recovery ratios at the head-end process of HTGR should be confirmed even at the laboratory level.

4. CONCLUSIONS

The fuel cycle scenarios for HTGR and the related back-end technologies were introduced, and subjects to improve the fuel cycle specifications are summarized. The representative scenario is reprocessing, and options are direct disposal, direct disposal with a plutonium burner named Clean Burn, and uranium based multi-recycling. Basically, the technologies are completed even though demonstration is needed in a FOAK plant. Finally, the subjects are arranged as follows:

- Reduction of a burden of graphite waste: determination of a reasonable near field model for public dose evaluation to simplify the disposal and reuse of graphite by annealing and refabrication to recover irradiation damage;
- Reduction of toxicity in uranium based multi-recycling: demonstration of recovery ratio more than 99.9% with the head-end of reprocessing.

With the completion of these subjects, it is expected that the HTGR fuel cycle will be improved and other SMR's fuel cycle with common technologies would be promoted.

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Paper ID#48

A COMPARISON STUDY ON THE BURNUP OF HTR-10 FUELS USING RADIOMETRIC AND MASS SPECTROMETRIC METHODS

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Extended Abstract

In the paper "A comparison study on the burnup of HTR-10 fuels using radiometric and mass spectrometric methods" published in PNE, burnup calculated from Cs-137 by radiometric method and Nd-148 by mass spectrometric method were compared. Results showed that the two values were not significantly different for fuels with a clear radiation history, but a significant difference was observed between Ft (total burnup) and F5 (fractional U-235 burnup) of Pebble 1, which was up to 2% FIMA. This result is much higher than expected and previously studied. Possible factors that may contribute to this disagreement in P1 is the inadequate recovery of uranium from irradiated particles.

In this paper, we optimized the leaching procedures by introducing oxidation at high temperatures and the extraction efficiency of the uranium content o UO_2 kernel has been significantly improved. Based on these results, the homogeneity of burn-values for HTR-10 is carefully and accurately investigated again.

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APPENDIX

MEETING PROGRAMME

A.1. MEETING ORGANIZATION

Chair	Ms Cécile Evans
Scientific Secretary	Ms Amparo González Espartero

A.2. METTING SESSIONS

The contributions to the meeting are classified into three major areas, and the meeting was divided info following three sessions:

SESSION I: IAEA Perspectives (Chaired by Mr Surik Bznuni)

Paper ID	Title	Presenter	Country	Org
ID#3	2022 IAEA SMR ARIS Booklet on Advances in SMR Technology Developments	Mr Hadid Subki	IAEA	NPTDS
ID#4	The IAEA Platform on SMR and their Applications: Progress and Achievements after One Year	Mr Stefano Monti	IAEA	NPTDS
ID#5	IAEA On-going Activities on Spent Fuel Management	Ms Amparo González Espartero	IAEA	NFCMS
ID#6	Information on the IAEA TM on Back End Opportunities and Challenges for Spent e-ATFs Management	Mr David Hambley	UK	NNL
ID#7	Consideration of Non-Water Cooled Reactors and SMRs in the IAEA Safety Standards	Ms Paula Calle	IAEA	NSNI
ID#8	Security Considerations for Back-End of Nuclear Fuel Cycle for SMRs	Mr Tariq Majeed	IAEA	NSNS
ID#9	Safeguards Considerations for SMR Fuel Cycles	Mr Kerrin Swan	IAEA	SGCP
ID#10	CPR on the Economic Appraisal of SMRs	Mr Saied Dardour	IAEA	PES
ID#11	Transport Safety and Back End of SMR Fuel Cycle	Ms Shazia Fayyaz	IAEA	NSRW

SESSION II: International Perspectives (Chaired by Mr Surik Bznuni)

Paper ID	Title	Presenter	Country	Org
ID#12	JRC Contribution to SMRs and the Back-End Fuel Cycle	Ms Concetta Fazio	EC	JRC
ID#13	Update on OECD/NEA Activities Regarding the Back end of SMR	Ms Rebecca Tadesse	OECD	NEA
ID#14	The Potential Impacts of SMRs on Multinational Cooperation at the Back end of the Fuel Cycle	Mr Charles McCombie	Switzerland	ERDO

SESSION III: Member State Perspectives (Chaired by Mr Andrea Salvatores, Mr David Hambley, and Ms Fatimah Al Momani)

Paper ID	Title	Presenter	Country	Org
ID#16-1	Safety Implications on the Back End of the Fuel Cycle for SMRs	Mr Surik Bznuni	Armenia	NRSC
ID#16-2	Important Aspects of Fuel Cycle to be considered during Development of New Nuclear Power Technologies in the Republic of Armenia	Mr Vahram Petrosyan	Armenia	Armenian Scientific-Research Institute for NPP Operation
ID#17	Fuel Cycle Considerations for WWR-SM Research Reactor	Mr Azizbek Boltabaev	Uzbekistan	The State Committee on Industrial Safety of the Republic of Uzbekistan
ID#18	Direct Recycling of SMR Spent Fuel for Uranium Utilization Improvement and Reduction of High-Level Nuclear Waste	Mr Nader M. A. Mohamed	Egypt	EAEA
ID#19	Design Extension Conditions for Spent Fuel Storage at the PWR NPP KRŠKO	Mr Tomi Živko	Slovenia	Slovenian Nuclear Safety Administration
ID#20	The Process of Decarbonization of the Domestic Power Industry in Poland using SMR Reactors	Ms Agnieszka Miskiewicz	Poland	INCT
ID#21	Considerations and Perceptions for Nuclear Fuel Cycle Back End Related to the SMRs under Consideration in Jordan	Ms Fatmah Al Momani	Jordan	JAEC
ID#22	Back End of the Fuel Cycle Considerations for Small Modular Reactors	Mr Andrea Salvatores	France	CEA
ID#23	Advanced Nuclear Reactors: What about the Back end? Focus on Treatment and Reprocessing/Recycling Aspects	Mr Renaud Liberge	France	Orano
ID#24	Molten Salt Reactor Technology	Mr Vincent Pascal	France	CEA

ID#25	Integration of MSRs in LW-SMR Fleets to Close their Fuel Cycle and/or Manage Waste	Ms Isabelle Morlaes	France	Orano
ID#26	Application of a Graded Approach to the Concept of Fuel Recycling	Mr Olivier Gregoire	Canada	Moltex Energy
ID#27	Finnish Perspectives on SMR Back End of the Fuel Cycle	Mr Timothy Schatz Mr Ville Koskinen	Finland	STUK, VTT
ID#32	The Anita Programme and SMR Spent Fuel Management from the Swedish Perspective	Mr Kyle Johnson	Sweden	Studsvik
ID#33	Challenges of Small Modular Reactor Used Fuel Management in Canada	Mr Blair Bromley Mr George Xu	Canada	CNL
ID#34	Overview of US DoE's Office of Spent Fuel and Waste Disposition Activities	Mr Jorge Narvaez Ms Natalia Saraeva Mr Stephen Kung	United States of America	DoE
ID#35	Progress on Considering the Back End of the Fuel Cycle for Small Modular Reactors	Mr Stuart Arm	United States of America	PNNL
ID#36	Investigations into Back End of the Nuclear Fuel Cycle (BENFC) Issues for Advanced Reactor (AR) Fuel and Accident Tolerant Fuels (ATF)	Mr Ramon Pulido	United States of America	SNL
ID#37	Evaluation of Advanced Reactor Spent Fuel Management Facility Deployment	Mr Milos Atz	United States of America	ANL
ID#38	Opportunities to Optimize Small Modular Reactor (SMR) Used Nuclear Fuel Management in the United States	Mr Rod McCullum	United States Of America	NEI
ID#41	Romania Strategy for Radioactive Waste and Spent Nuclear Fuel according with the Development of the Expending Nuclear Programme	Ms Alice Mariana Dima	Romania	Nuclear and Radioactive Waste Agency
ID#42	Challenges for Transport and Storage	Mr Brut Stéphane	France	Orano
ID#43	Transport Considerations for SMR Fuel Cycle and TNPPs	Mr George Burnett	UK	NTS
ID#44	Back End the Fuel Cycle Considerations for CAREM the Argentinian Small Modular Reactor	Ms Laura Kniznik	Argentina	CNEA
ID#45	Managing Fuel from SMRs: UK Framework and HTGR Gap Analysis	MR David Hambley	UK	NNL
ID#46	TRISO Fuel Management Depending on the Choice of the Fuel Cycle Research Currently Conducted at INCT, Warsaw, Poland	Ms Katarzyna Kiegiel	Poland	INCT

ID#47	Fuel Cycle Scenarios and Back End Technologies of HTGR in Japan	Mr Yuji Fukaya	Japan	JAEA
ID#48	Deconsolidation of HTR-10 Irradiated Fuels & Measurement of Burn up for SFE Storage	Ms Xiaotong Chen	China	Tsinghua University

A.3. BREAK OUT SESSIONS

Group Discussion	*Title*	*Chairs*
Group Discussion I	Light Water Reactor (LWR) SMR designs	Mr Timothy Schatz
Group Discussion II	Hight Temperature Gas-Cooled Reactor (HTGR) SMR designs	Mr David Hambley
Group Discussion III	Advanced Reactors and Molten Salt Reactor (MSR) SMR designs	Mr Andrew Worrall

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