Management of Spent Fuel from Nuclear Power Reactors

Learning from the Past, Enabling the Future

Proceedings of an International Conference
Vienna, Austria, 24–28 June 2019

OECD NEA NUCLEAR ENERGY AGENCY
European Commission
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IAEA
International Atomic Energy Agency

MANAGEMENT OF SPENT FUEL FROM NUCLEAR POWER REACTORS

The Agency's Statute was approved on 23 October 1956 by the Conference on the Statute of the IAEA held at United Nations Headquarters, New York; it entered into force on 29 July 1957. The Headquarters of the Agency are situated in Vienna. Its principal objective is "to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world''.

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MANAGEMENT OF SPENT FUEL FROM NUCLEAR POWER REACTORS

LEARNING FROM THE PAST, ENABLING THE FUTURE

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IN COOPERATION WITH THE
OECD NUCLEAR ENERGY AGENCY,
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International Atomic Energy Agency
Vienna International Centre
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FOREWORD

Nuclear power can help address the twin challenges of ensuring reliable energy supplies and curbing greenhouse gas emissions. The 441 nuclear power reactors in operation in 30 countries today supply over a tenth of the world's total electricity and a third of all low carbon power. Nuclear power will continue to play a key role in the world's low carbon energy mix for decades to come.

The safe, secure and sustainable management of spent fuel from nuclear power reactors is key to the future of nuclear energy. It is a complex undertaking, covering many technological aspects related to the storage, transportation and disposal of the spent fuel and the high level waste generated from recycling through its reprocessing. Furthermore, research and development has established the feasibility of advanced processes, such as partitioning and transmutation, which have the potential to further reduce the impact of nuclear waste and preserve natural resources. The implementation of any selected strategy can take decades, and national strategies need to be flexible enough to make it possible to accommodate potential future options and new technologies that will enhance and improve the safety and sustainability of nuclear power. Allocating the necessary resources to implement the strategy is often difficult.

There is a lack of visibility regarding spent fuel storage durations, partly due to the long lead time required to develop a deep geological repository, which subsequently impacts the handling and transportation of spent fuel in the long term. This also extends to next generations to ensure the availability of future technologies, including possible advanced recycling options, underground disposal facilities and suitable financial, regulatory and political frameworks. It is paramount to take an integrated view of the nuclear fuel cycle to ensure that influences from, and impacts on, all stages of the nuclear fuel cycle are clearly identified and understood, enabling effective decision making in the back end of the fuel cycle to guarantee efficient, safe and secure management of the generated spent fuel.

In 2019, the IAEA organized the International Conference on the Management of Spent Fuel from Nuclear Power Reactors: Learning from the Past, Enabling the Future, the latest in a series of conferences on the topic. The purpose of the conference was to provide a forum for the exchange of information on national spent fuel management strategies, on the ways in which a changing energy mix could influence these strategies and on how they support the achievement of national energy goals. Following the theme of the 2015 conference, the 2019 conference aimed to illustrate the positive impacts that an integrated approach to the back end of the nuclear fuel cycle has on the management of spent fuel, anticipating the impacts that new developments in fuel design and operation may have on back end activities. It considered the latest technological developments, as well as regulatory requirements and safety aspects.

The conference also allowed for the evaluation of advances in the management of spent fuel to overcome current issues and the identification of expected future challenges and possible strategies for dealing with them.

This publication provides a summary of the different conference sessions, including the full text of invited papers and of the opening and closing speeches delivered during the conference. The papers and posters presented at the conference are available in the on-line supplementary files.

The IAEA would like to express its appreciation to the members of the International Scientific Programme Committee, the Secretariat of the Conference and the IAEA staff supporting it for their commitment, professionalism and engagement. The IAEA officers responsible for this publication were A. González-Espartero of the Division of Nuclear Fuel Cycle and Waste Technology and G. Bruno of the Division of Radiation, Transport and Waste Safety.

EDITORIAL NOTE

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1. EXECUTIVE SUMMARY AND MAIN CONCLUSIONS

The safe, secure, reliable and economic management of spent fuel (SF) arising from nuclear power reactors is key for the sustainable utilization of nuclear energy and covers many technological aspects related to the storage, transportation and disposal of the spent fuel and the high level waste (HLW) generated from recycling through its reprocessing.

The sustainability of nuclear energy involves the preservation of natural resources and the minimization of generated wastes. In some countries, the remaining uranium (U) and plutonium (Pu) are currently industrially recovered from spent fuel and recycled as mixed oxide (MOX) in thermal reactors, saving natural uranium resources and generating vitrified HLW and irradiated spent MOX fuel. Future advanced fuel cycles based on the multirecycling of U and Pu in thermal reactors in the short term and in Gen-IV reactors in the longer term will allow nuclear energy to be almost independent of uranium natural resources and to dramatically reduce the generated wastes in terms of heat loading, radiotoxicity and proliferation risks.

The last IAEA International Conference on Management of Spent Fuel from Nuclear Power Reactors, held in June 2015, highlighted that there is little integration in the fuel cycle in terms of analysing how decisions made in one part of the fuel cycle may affect another part. Introducing efficiencies into the individual steps in isolation can create additional challenges in subsequent steps. A worst-case scenario would be that a decision taken today forecloses a transition to another step tomorrow. In most cases a technical solution can be found, but this is likely to come at a price (cost, resource utilization, etc.). Therefore, one of the main challenges is to maintain enough flexibility to accommodate the range of potential future options for the management of spent fuel as well as to define and address the relevant issues in storage and transportation, given the current uncertainties regarding the storage duration, the availability of future technologies and future financial, regulatory and political conditions.

In this context, the IAEA organized the International Conference on the Management of Spent Fuel from Power Reactors, in Vienna, from 24 to 28 June 2019, with the theme “Learning from the Past, Enabling the Future”. Special attention was given to the Young Generation of professionals to support bridging the gap with the current ageing industry workforce. Support for the development of young professionals in the nuclear sector is essential in the promotion and continuity of a safe and sustainable nuclear power.

The broad scope of the conference covered all stages of the management of spent fuel from nuclear power reactors from the past, present and future technologies, and how it can be affected by the decisions taken in the rest of the nuclear fuel cycle. With this vision, the Conference was structured in seven tracks covering national strategies, spent fuel and HLW storage and subsequent transportability, transportation in the backend, recycling as a spent fuel management option, impacts of advanced nuclear energy systems on the backend, disposal of spent fuel and HLW and challenges in an integrated approach for the backend. This is reflected in this publication, with a summary of all presented papers and the invited papers in full for each track. All presented papers can be found in the supplementary files on the IAEA publication website.

A number of approaches to spent fuel management have been and will be adopted around the world due to differences in the adopted technologies and organizational arrangements which, in turn, arise from a range of technical and societal factors. There is a clear consensus that spent fuel management must encompass all activities from discharge from the reactor core to emplacement of fuel and/or waste in a disposal facility.

Embarking countries are consciously addressing backend management before building nuclear power plants and there is evidence of an intent to ensure disposal facility development is progressed alongside the development of power plants to provide comprehensive backend solutions before the end of power generation. Such plans are technologically feasible and address intergenerational equity issues faced by many mature nuclear countries.

There is growing evidence that a combination of consistent policy, effective public engagement and education, strong regulation and commercially led delivery provides a sound approach for an effective delivery of a comprehensive spent fuel management system. There is evidence from many countries, especially embarking countries, of learning lesson from the past and developing robust, integrated strategies for managing spent fuel from discharge until all wastes are disposed of.

There is evidence of a trend towards delaying implementation of fast reactor-based closed fuel cycles due to economic conditions arising from the ready availability of recoverable natural uranium, although this remains an aim for the future in many countries with a strong recycling programme. Nevertheless, necessary development of fast reactors (FRs) and associated fuel cycles continues and provides a valuable role in developing the technologies that would support future deployment. In the meantime, development of fuel and fuel cycles that enable multi-recycling in thermal reactors is being pursued mainly by France and the Russian Federation to achieve benefits in resource use and environmental impact beyond the current mono-recycling practice.

A number of national strategies reflect the need to make available sufficient spent fuel storage capacity to bridge the gap between the generation of spent fuel and the foreseen commissioning and operation of deep geological disposal facilities. The industry continues to develop safe technologies for longer term fuel storage. However, a number of such systems, particularly large capacity canister-based systems, will not be compatible with current disposal concepts, so there is a need for urgency on work to understand optimization of the whole backend and to actively implement these strategies on the ground.

The current activities worldwide on interim and long-term storage associated with both wet storage and dry storage systems were addressed. Discussions were conducted regarding the implementation of ageing management programmes to ensure fuel integrity and identify degradation mechanisms. Efforts to ensure the safety and security of spent fuel and high-level wastes were discussed and specific features that support safety and security in wet and dry storage applications were included. Furthermore, discussions on data collection and testing activities to improve the characterization of spent fuel and high-level waste to improve and support storage and transportability were conducted. At the end of the day, any component and material in spent fuel storage systems will degrade. Therefore, ageing management programmes with monitoring and inspection are essential for long-term storage based on degradation mechanisms coupled with operating experience and lessons learned. In this context, valuable presentations and discussions were made during the Conference.

Regarding transportation in the backend, experiences in the USA with planning for transportation following the shutdown of a site were addressed, with discussions on the importance of preparation activities just before the site closure and immediately after. An important lesson learned is the building of relationships with the people on the site for information.

Thermal output is a limiting factor for fuel transition from wet to dry, especially in the case of higher burnup fuels. Various modelling approaches were discussed for comparison of conservative modelling results against realistic modelling results. The experience with the transportation of spent fuel in France was highlighted, with Orano conducting around 200 shipments per year. Efforts taken in the USA to support and prepare transportation of high burnup fuels was discussed, including considerations on radionuclide inventory, internal pressures and cladding performance. Extensive tests of high burnup fuel that has undergone hydride reorientation and their impact on the cladding stress was discussed. Challenges with the dry storage and transportation of high burnup and damaged fuels in Spain were also discussed. Currently, transport of spent fuel in Spain is limited to less than 45 GWd/tHM; however, there are efforts underway to remove this limitation. Disposal of fuel in Finland is expected to begin in the 2020s and preparations for the transportation of spent fuel to the disposal facility are ongoing.

The experience in Russia of international multimodal transport of spent nuclear fuel, which includes research reactor spent fuel from 13 countries, was highlighted. To support shipments, a special semi-trailer was designed. The packages necessary to support road, rail, air and sea transport were discussed after the presentations. The USA highlighted a collaborative international multimodal spent nuclear fuel transportation trial utilizing three surrogate PWR assemblies from Spain to the USA. Data were collected during all modes of transportation. Real transport data collected was compared with data from 125 tests and showed that the tests are bounding. The experiment concluded that handling activities provided the biggest strain on the fuel. Additionally, the evolution of transport regulations for spent fuel was discussed, including the challenges of implementing the regulations. One of the most significant changes occurred in 1964 when mechanical test requirements were introduced. Future challenges include demonstrating compliance for fuel designs for longer reactor cycle times and higher burnups and maintaining transportability with the trend towards the use of dual-purpose casks. Experience with the transportation of sensitive nuclear materials and spent nuclear fuel in the UK was discussed. It was noted that there has never been a release of nuclear materials during the performed domestic or international transports. The design features and licensing criteria for the International Nuclear Services (INS) ship vessels used in the transport was presented in the Conference. The UK has a comprehensive approach for the security of transport of nuclear materials in both domestic and international transports.

Countries reprocessing spent fuel are continually improving existing mature technologies. The French La Hague reprocessing plant and the Russian reprocessing plant (RT-1) PO “Mayak” are increasing the range of LWRs and Russian Reactors spent fuels, including damaged fuels, to be treated and performing investments to both increase competitiveness and secure long-term operations. This is leading to continuous development and implementation of new technologies. In La Hague plant, these improvements include the implementation of a Cold Crucible Induction Melter, R&D performed regarding clogging issues in dissolution and separation steps, corrosion issues in evaporators and vitrification steps in the reprocessing technological scheme. The Russian reprocessing plant RT-1 presented plans for increasing the capacity of reprocessing from 400 to 600 tHM/year by 2022, with upgrading the technology, including new cutting machine and a voloxidation unit for tritium removal from effluent releases.

Part of the presentations on spent fuel reprocessing were focused on the status of existing Pu mono-recycling technology using MOX fuel (experience of MOX fuel supply and MOX fuel performance enhancement) and the development of multi-recycling technologies for the existing fleet of LWRs (several types of REMIX processes for multi recycling of RepU and Pu in the Russian Federation and CORAIL and MIX processes for Pu multi recycling in France). These recycling options can reduce the need for natural uranium from 25% to 35% compared to the open fuel cycle and can provide a sustainable solution for the transitioning period from once-through recycling (currently implemented cycle) to a fully closed fuel cycle with FRs. The preliminary assessments of the MIX and CORAIL concepts show that they are capable of recycling spent MOX and ERU (Enriched Reprocessed Uranium) fuel, and of stabilizing the spent fuel and plutonium inventories. The Russian Federation studies showed the economic benefits for REMIX technology in the closed fuel cycle in comparison with the open fuel cycle.

Japan continues to engage in the development of advanced recycling technologies aimed at the minimization of generated waste burden as a target for the reduction of geological disposal footprint. The impact of spent fuel characteristics (fuel burnup, spent fuel cooling period) and radioactive waste characteristics (waste loading in vitrified waste, separation of minor actinides as heat-generating nuclides) was discussed.

Aspects of advanced reactor system development in terms of reducing the impact of nuclear power on the environment were discussed during the Conference, highlighting the importance of international collaboration in this regard.

The report "Back End Fuel Cycle Strategies in uncertain Generation-IV futures" summarizes some multilateral studies on the impact of a combination of different reactor systems to achieve a harmonious nuclear closed fuel cycle, in particular, the main findings of the SYNERGIES collaborative project completed in the framework of INPRO. The European R&D project GENIORS, focused on the development of efficient systems for plutonium multi-recycling and minor actinides recycling as MOX fuel in Gen-IV reactors, was described in detail, including the importance of Technology Readiness Level (TRL) of both reactors and associated fuel cycles.

The national R&D programmes of Japan, the Russian Federation and India highlighted the development of advanced fuel cycles with new reactor systems (fast reactor with sodium or lead coolant and accelerator-driven systems (ADS)) and new approaches to actinide recycling in molten salt reactors. The specific possibilities of industrial and medical uses of fission products contained in spent fuel were noted. The reported results show a promising opportunity to apply advanced technologies to create effective nuclear fuel cycles with a reduction of the amount of radioactive waste.

The introduction of Advanced Technology Fuels (ATF) and the need for consideration of the management of spent ATF fuels from the design phase, including storage, recycling and disposal, aroused great interest with the participants.

An overview of the issues related to disposal: siting, multinational approaches, cost and knowledge management was given. Final disposal of SF and HLW have been analysed in terms of multiple nuclear fuel cycle options. The French final disposal programme continues an ongoing industrial project in its design phase, aiming at starting the inactive phase around 2030. Possible financing approaches of a multinational repository were discussed. The effort will be pursued with the view to engaging service providers and potential customers. One theme particularly highlighted by Canada and Finland was the vital and essential role that local stakeholders might play. Any future disposal programme should take these socio-political factors into account from the beginning.

The importance of the integration of the various stakeholders involved in the implementation of spent fuel management programmes from the beginning was highlighted, as e.g. nuclear power plant operators, spent fuel management service providers, waste management organizations, safety authorities and technical support organizations (TSOs), R&D entities as well as public communities and administrations. Maintaining this network while implementing activities is essential.

Modelling and simulation analysis required to assess various spent fuel management options, including the identification of risks and opportunities, were discussed. These analyses will allow to define mitigation and optimise spent fuel management strategies.

Looking at the current global spent fuel inventory and its future growth, and associated uncertainties with the need to implement extended spent fuel storage periods, innovative methodologies, integration of risks and valuing flexibility, were thoroughly discussed. Some, for instance, allow for the development of optimal portfolio management of spent fuel inventories considering all options (direct disposal or recycling options) thus minimizing financial risks. Other methodologies include these uncertainties by design of the spent fuel management programmes.

Although spent fuel storage for extended periods is now a reality shared by various stakeholders on a worldwide basis, it needs to be accounted for, and pursuing the development of an end point, i.e. geological disposal of spent fuel or HLW from reprocessing, was recognized as a key enabler or even a must by all participants to ensure the sustainability of nuclear power.

Finally, collaborative work on international/multinational management schemes based on the development of shared infrastructures for storage, reprocessing/recycling and disposal was also described as an effort to pursue to overcome challenges in spent fuel management system implementation.

MAIN CONCLUSIONS

1. The value of enhancing and fostering international collaborations, for sharing experiences and lessons learned was recognised.

2. There is evidence from many countries, especially embarking countries, of learning lessons from the past and developing robust, integrated strategies for managing spent fuel from discharge until all wastes are disposed of.

3. The IAEA plays an important role in providing opportunities for embarking countries to learn from mature programmes.

4. Recycling of spent fuel continues to play an important role and there is a focus on developing multi-recycling technologies to be applied using thermal reactors that can provide a sustainable solution for the transitioning period from once-through recycling (currently implemented industrial cycle) to a fully closed fuel cycle with fast reactors. There are examples of countries considering innovative recycling technologies to reduce the

burden of generated wastes and footprint of disposal facility, by recycling long-lived products for medical applications.

5. Implementation of fast reactor based closed fuel cycles is being delayed in many countries, although research and development activities on large reactors and associated fuel cycles continue in some countries.

6. A number of national strategies reflect the need to make available sufficient spent fuel storage capacity to bridge the gap between the generation of spent fuel and the foreseen commissioning and operation of deep geological disposal facilities. There is evidence of greater attention being given to impacts of fuel cycle on disposal and vice versa, especially with uncertainties on the requirements and acceptance criteria of the disposal facilities. There is an urgency on working to understand and optimise the whole backend and to actively implement these strategies on the ground.

7. Costs and a lack of sustainable funding is a concern for many countries. Costs and associated risks and uncertainties were explored during the Conference.

8. It was recognised the importance to invest resources in R&D to address future challenges (e.g. as the management of spent ATF or the incorporation of safeguards into facility designs)

9. The current industrial working force is ageing and there is an urgent need for knowledge management, records preservation and efforts in developing the young generation to continue safe, secure and sustainable fuel cycle management.

10. The importance of the integration of the various stakeholders involved in the implementation of the spent fuel management programmes from the beginning and maintaining this network while implementing activities is essential. Public understanding is paramount and key as a first driver and second step must be politics as decision makers.

2. OPENING SESSION

2.1. CONFERENCE CHAIRWOMAN'S OPENING REMARKS

Opening speech as provided, verbatim.

Susan Y. Pickering
Director Emeritus, Sandia National Laboratories, USA

Good morning and welcome to the 2019 International Conference on Management of Spent Fuel from Nuclear Power Reactors. I am delighted to be chairing the conference and look forward to spending the week with you.

I would like to acknowledge the contributions made by the Programme Committee, IAEA Secretariat, and all the other people who worked very hard and for many months to bring forth this conference. There is no better team!

The theme we chose for the week is "Learning from the Past, Enabling the Future." Each country is on its own journey with nuclear power and has its own past and future. Likewise, each individual is on his or her own journey. Whether we come from mature or emerging nuclear programs, all of us can learn from each other. This conference provides an ideal venue for these learnings to occur.

We thought deeply about the theme for the conference and the type of information that would be most useful to you. We designed the tracks with a specific logic in mind – we cover high level national strategies, the storage, transportation, recycling, and disposal of spent fuel; including understanding the impacts of advanced designs on spent fuel management; and have a session about how all of these impact each other.

It is an exciting time for nuclear power. As of February this year, about 11% of the world's electricity is generated by about 450 reactors. Since 2015, over 25 GW(e) have been added globally. Another 33 GW(e) are expected to be on line by 2020. Thirty countries have operational nuclear power plants. About 60 new reactors are under construction. Asia, especially China, has the newest construction; followed by Eastern Europe and Russia.

In my country, the United States, nuclear energy provides over 55% of the carbon-free generating capacity. That results in a reduction of 528 million metric tons of CO_2 per year!

At this year's International Congress on Advances in Nuclear Power Plants over 40 nuclear associations signed the Declaration of Clean Energy which called for a doubling of public expenditures on nuclear research and development. The intent is to allow nuclear energy to make it contribution towards carbon-free energy.

Every nuclear power programme must manage spent fuel. The management of spent fuel will require commitments of resources spanning decades, possible spanning centuries. There will be many aspects of spent fuel management that must be addressed, both technical and non-technical; including safety, security, economics, political, legal and regulatory, and societal.

Maintaining a high level of operational excellence will be difficult over the long lifespan of a nuclear facility. Pressure to reduce costs can lead to unwise decisions. Personnel and organizational turnover can lead to lost knowledge. Complacency could grow over time. Facilities age and could become less reliable. New, unanticipated vulnerabilities could emerge over the years, such as cyber security.

Nuclear systems are often perceived as controversial. Stakeholders are many, often have opposing views, and may be a source of conflict. The impact of stakeholders must be appreciated as they may influence policy and decision makers. Stakeholders generally want frequent engagement, transparency, and influence. The relationship between a nuclear facility and its stakeholders is important, and resources must be applied to support it. Collaborating with the public, stakeholders, and local governments increases the likelihood of success.

An understanding of risk is critical to properly managing a nuclear programme. Even though accident frequency estimates are extremely low, consequences could be significant, costly, and long-lasting. The systems are complex and require credible science and sophisticated engineering to ensure risks are managed properly. Technically competent leadership in the government sponsor, regulatory agency, and implementing team is a major success factor.

Leaders at all levels in an organization must embrace the behaviours that foster a strong nuclear safety culture. They must accept that there will be surprises, and plan for normal and abnormal events. They must understand uncertainty, risk, margin, defence-in-depth, and resiliency. Competent people are the most important success factor for a strong, safety culture. As Admiral H.G. Rickover, the father of nuclear safety in the USA, said, "*Rules are not a substitute for rational thought*."

There are many directions a nuclear programme could take. A country may choose an open fuel cycle and directly dispose of spent fuel. It may choose a closed fuel cycle and reprocess spent fuel. Or it may choose a hybrid model and do both. A county may implement fuel leasing where spent fuel is returned to the supplier residing in another country. A country may be a partner in an internationally shared disposal facility. For every possible direction a country could choose, there are lessons to be shared.

Nuclear power systems are complex and integrated. We need to view these systems from the cradle to the grave. Nuclear power has over 17 000 reactor-years of experience! What a wealth of knowledge and expertise to help ensure nuclear power is safe and secure. Throughout the week will learn from this experience base and explore issues and successes, so we may apply these learnings and build a better future.

Thank you!

2.2. INTERNATIONAL ATOMIC ENERGY AGENCY DEPUTY DIRECTOR GENERAL – NUCLEAR ENERGY'S OPENING REMARKS

Opening speech as provided, verbatim.

Mikhail Chudakov
Deputy Director General, Head of the Department of Nuclear Energy
International Atomic Energy Agency

Thank you, Dear Susan, Chairwoman of the Conference.

Good morning, Ladies and Gentlemen.

I am pleased to welcome you to this *International Conference on the Management of Spent Fuel from Nuclear Power Reactors*.

In September last year, we devoted our annual Scientific Forum on *Nuclear Technology for Climate: Mitigation, Monitoring and Adaptation*. This October, we are organizing an *International Conference on Climate Change and the Role of Nuclear Power*. These actions reflect the need to fight against climate change, recognized as a global priority in the Sustainable Development Goals and the Paris Agreement.

Nuclear technologies have an important role to play in mitigating greenhouse gas emissions, in monitoring the effects of climate change and in adapting to them. At present, nuclear power produces about 10 percent of the world's electricity, but it generates almost one third of the global total of low carbon electricity.

However, for nuclear power to be sustainable, the safe, secure, reliable and efficient management of its fuel cycle is paramount, in particular the management of the spent fuel and radioactive wastes generated. It is a complex undertaking, involving storage, transportation, recycling and disposal steps. This challenge is as much for policymakers as for engineers. Indeed, technical solutions for the management of spent fuel exist – whether reprocessing and recycling, or conditioning for spent fuel disposal in deep underground repositories. However, the implementation of any of these options can take decades, and allocation of the necessary resources is often challenging.

Ladies and Gentlemen,

The theme of this year's conference is *Learning from the Past, Enabling the Future*. It brings together Member States with decades of nuclear power operating experience and countries that are developing or considering a nuclear power programme.

There is progress in the implementation of strategies for spent fuel management. Repository operations expected in the Onkalo facility in Finland, in Forsmark in Sweden and in the CIGEO facility in France provide confidence that it is likely that all steps of the back end of the fuel cycle are being or will be demonstrated in the next decade or so. At the same time, there are countries where strategic decision making and implementation have stalled or remain slow.

The timeframe in fully implementing the strategy, coupled with the trend of increasing nuclear power capacity, are leading to an increase in the quantity of spent fuel accumulating in storage for long periods, in some cases beyond originally licensed duration.

Research, development, engineering and demonstration activities are carried out in Members States to address nuclear power challenges and to enhance safety and security when both establishing and implementing strategies for managing the back end of the fuel cycle. Much can be gained from sharing knowledge, experiences, lessons learned and best practices.

In this regard, the IAEA has been conducting a series of Coordinated Research Projects focusing on the behaviour and performance of spent fuel and systems, structures and components, under storage conditions. A technical report, which compiles operational experience and results of research and development accumulated in the framework of six such projects since 1981 has recently been published. A new project on ageing management programmes for dry storage systems is underway.

Another mechanism for disseminating knowledge and experience is the IAEA's range of topical networks, such as on Spent Fuel Management, or on Underground Research Facilities. And, the IAEA has also published new and revised safety standards, such as the Revision of the Regulations for the Safe Transport of Radioactive Material.

Ladies and Gentlemen,

The scope of this year's conference is very broad. It includes all steps for managing spent fuel once it is discharged from the reactor core. It covers both policy aspects as well as technical topics, such as storage and current recycling technologies, as well as innovative systems to improve sustainability, safety and security of nuclear power. It also builds up on the theme of the last Conference held in 2015 and includes two sessions dedicated to integrated views of the nuclear fuel cycle to ensure that influences and impacts from decisions made at all stages of the nuclear fuel cycle are clearly identified and understood. We hope this helps enable effective decision making in the back end of the fuel cycle for efficient, sustainable, safe and secure management of spent fuel.

The importance of having the right scientific, technical and engineering skills and maintaining these competencies go hand in hand with ensuring ongoing safety and delivering comprehensive and safe management of spent fuel. The nuclear industry is already seeing an influx of young professionals and the need to support their development going forward is essential for safe and sustainable nuclear power. That is why we give special attention to the young generation at this conference: As you know, we have organized a Young Generation Challenge, and we will give young professionals from around the world an opportunity to co-chair a technical session with a senior professional and to deliver an oral presentation.

Ladies and Gentlemen,

More than 300 experts from over 50 Member States and 8 International Organizations are attending this conference. There will be more than 70 oral and over 60 poster presentations. I would like to thank you all for your contributions. I especially thank our Chairwoman and all the experts who have actively supported the organization of this important conference.

I wish you every success and look forward to learning about the outcome. Thank you.

2.3. DIRECTOR GENERAL ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT – NUCLEAR ENERGY AGENCY'S OPENING REMARKS

Opening speech as provided, verbatim.

William D. Magwood, IV

Director General of the OECD Nuclear Energy Agency

Good morning.

Let me begin by thanking Mr Chudakov, who has been a good partner for with us for many years. We had one of our semi-annual coordination meetings just last week as a matter of fact. And I thank the rest of the IAEA team for having me here today. It's always a pleasure to coordinate with the Agency. Also, as I look around the room, I see many people here who have been at various NEA meetings, so I know we have a lot of cross fertilization, a lot of interaction.

Conferences like this are very important. NEA conferences often focus more narrowly on expert-level issues so we don't generally hold broad technical conferences like this so it's very important to bring the whole community together to talk about issues in a cross-cutting fashion, so I appreciate the IAEA's continued dedication to holding forums like this and again thank you for having me join you.

The theme today is interesting for me, learning from the past. I was reflecting this morning as I thought about that theme, that perhaps I am the past. Because it occurred to me that I've started my first full-time job, 35 years ago this month working on, believe it or not, thermal analysis for Yucca mountain and monitored retrieval storage systems. I must've done a fantastic job because just see how much progress has been made since then. But it goes to show how much time many of us have spent on this area. There are those of us in this room, I suspect, that've spent most of our careers talking about this subject in one form or the other over the years. And I'm sure there's many people who feel that the lack of progress has been very frustrating. But at the same time, I think it's important to recognize that these issues are particularly complex from a social perspective. I'll talk about that again in a moment.

I think now most of you know about the NEA, so I'll briefly note that we have 33 Member Countries now, a couple years ago we had two new countries joining us, Argentina and Romania, we're pleased and very happy to have them with us, they add a lot to our conversations. We've added a new standing technical committee in the area of decommissioning and legacy management last year. But at the same time, the focus has always been on bringing the countries with deepest experience in nuclear together to try to solve difficult problems and we try to provide a framework to do that.

Obviously one of the most difficult problems we face today is dealing with the issue of climate change. I'm sure many of you have seen this picture thousands of times. I think it's good to celebrate the accomplishments of COP21 and the COPs that have come afterwards. But these accomplishments I think sometimes get underplayed in their difficulty.

I'd like to point to a chart, which the OECD issued about a year ago, and I draw your attention to the last line where it says that in order to meet COP21 objectives, annual emissions from electricity we need to decline 73% globally and 85% in OECD countries. And the more you know about how large and immense the electricity infrastructures are in OECD countries, the idea that they're going to reduce emissions by 85% over the next few decades, it really brings home the difficulty of this challenge. The question is: how are we going to do this?

You know the International Energy Agency, IEA, our sister agency of the OECD framework, did this analysis here a few years ago and it basically said we have to do a lot of everything. We will need carbon sequestration. We have to use a lot of wind, a lot of solar, we have to be very efficient. And at the same time, we have to more than double our nuclear capacity globally if we're going to meet these targets. And I think it's fair to say that in almost every one of these categories we're falling short today. Particularly in the area of nuclear.

Many countries have put big emphases on renewables and we do expect to see a large deployment of variable renewable energy in next few decades. I think that's natural and I think that there's a lot of benefits of renewables for many countries. The NEA has done a lot of technical analysis and economic analysis on this. Our analysis highlights that we're not going to see a one size fits all approach that will work for every country around the world. We're going to see that for some countries with the right kind of resources that it'll make sense to have higher proportions of renewables, solar or wind, depending on the circumstances. But for others it won't. Every country really needs to analyse very closely how it's going to meet its obligations and not assume that just because someone sets an arbitrary target that they can meet that target in an economic fashion. And whatever your resource is, when you reach that 40 to 50% level of variable renewables the ability to control the grid in a reliable fashion comes into challenge. And there's a lot of experts that tell me, grid experts, that they really don't know how to manage a grid with higher loads in renewables in that. At least not the large grids that are in most OECD countries.

While these big investments in renewables are proceeding, emissions have continued to increase. As a matter of fact, IEA notes that last year we had highest emissions globally of CO_2 that we've ever had. So, despite the big investment in renewables, despite the political focus on climate change, we aren't making enough progress. So, I think this does present an opportunity for nuclear energy to reassert itself. There's a lot of discussion about small reactors, about Gen IV reactors and there's a lot of energy in a lot of places to push forward those technologies.

And we're going to have to see these technologies be successful if nuclear is going to be more cost effective and flexible, and if nuclear is going to be able to ensure very high levels of safety at low cost. At the same time, we have to deal with the fuel cycle and nuclear waste. And this issue of "how does nuclear fit into the evolving framework" is one in which that we spend a lot of time at the NEA. This is very challenging. You know nuclear energy grew up at a time when there were very high energy prices. It grew up at a time when large institutions, large electric utilities had a lot of leeway, had a lot of resources. And now we're moving into a period where there's less money in the electric business, less resources for research and less resources in terms of personnel. So how do we fit in that new framework?

One of the big issues obviously is going to be is the issue you're here to talk about – high level waste. The public expects us to deal with high level waste. The public expects that we should know what to do with the materials that we're generating in our nuclear power plants. And we have quite a bit it around; we have about 300 000 tonnes that we've been regenerating in NEA countries over the last several decades.

But at the same time, I hasten to note that this isn't a crisis. We have the time to do this the right way. We know how to store nuclear waste very safely. On-site storage and interim storage can last for decades and some places, some countries are expecting to store nuclear wastes for a hundred years. You can argue about the wisdom of waiting that long, but the truth is that you can do it. You can do it cost effectively, you can do it safely, there's no reason to rush. And others are looking at reprocessing and you'll be talking about that in this conference both conventional reprocessing, using PUREX technology, but also advanced reprocessing. And I personally believe that advanced reprocessing should be something that is very aggressively pursued around the world. We should spend more resources on that than we are today. But whatever you do, whatever approach you take, at the end of the day, you're very likely still going to need some kind of deep geological repository to dispose of whatever's left. It's very, very difficult to imagine any technical system where everything disappears. So, I think you are always going to need some kind of deep geological repository.

So, it makes sense to go forward to developing deep geological repositories. We have a lot of faith in this area of technology, we've spent decades analysing this around the world within the NEA framework. We have done exhaustive analysis, exhaustive studies looking at deep geological repositories. and we have a lot of faith in this.

We have a lot of faith in it because the years of research provides great confidence that deep geological repositories provide multiple barriers of protection to isolate these materials from the environment and from humanity. And that we're very confident, that we can build these facilities in a very effective manner.

That said, there are not that many countries that made as much progress as we would like. There are some notable exceptions and I think these chart highlights the most important ones, and particularly, I look at Finland. And you know that Finns, I think are really the global heroes of this area, because they will be the first ones to license and operate a deep geological facility. I'm very confident that it'll happen, and it'll happen around the time frame of 2023. Once that happens it will show the rest of the world that this is not an insurmountable technical problem and it's not a science problem, it's a political problem. It's a problem of finding out how do you site a repository and not an issue of technology. We know how to do this. And the Finns will show the rest of the world that they can do it, and if the Finns can do it, I'm sure the other countries can as well

Now that's not to say there aren't significant policy level issues that have to be addressed. Reaching consensus on retrievability is something that is debated in many countries. Some countries have settled this issue, other countries have not. Globally there's not a good consensus on this. This is something we probably need to talk about. Models for human intrusion are very difficult and that could be challenging when you go through the regulatory processes. But probably one of the most difficult things is maintaining policy stability. In many countries you can have a decision on one decade and next decade it gets reversed and you move in different direction It's can be very difficult to maintain a consensus for decades and decades and building a deep geological repository isn't something that happens in a few years, it takes decades of commitment. And if you change policies every 10 years, you're never going to get it done. So that's very difficult. But the most difficult issue I think that most countries deal with is simply the site. Finding a public policy process that enables you to identify a site and to convince the local population and the communities along transportation routes to accept that a repository site is appropriate. So, this is the big challenge that I think we have.

At the NEA we have several different approaches to try to help with this. One that's been very productive over the last few years has been the Forum on Stakeholder Confidence, some of you I think have participated in these activities. This is a group that works together to compare notes on “how do you work with the public?”, “how do you involve stakeholders in these conversations”, “what are the processes that worked most effectively over the years”. And we also hold these national workshops where international experts go to a country that’s looking at a siting process to give the local people some confidence that what their government is telling them, the processes being used, is very much like what’s being used in other countries giving them confidence that what they’re being told is very consistent with the practice around the world. And we’ve found that this has been very helpful. The last one we held, which was in Switzerland, involved young people just as your conference does today. And we think bringing young people, not just young people in terms of 30-year-olds and 20-year-olds, but even the teenagers, into the conversations is very important. Because after all they’re the ones who’s going to have to finish this work and I think it’s important that they understand what we’re trying to accomplish.

We are working on a new initiative which was launched by the Government of Japan during the G20 Energy and Environment Ministerial just last weekend. This effort is to assemble a high level round table on radioactive waste management. The first meeting of this high level round table is going to be held in October at the NEA. 13 countries have joined this so far, Japan will be joined by France, United States, China, Russia, Germany, Finland, and several other countries. And these countries will send very high level people to work on a round table to talk about how we accelerate this work going forward, what are the policy level things that we can do to help, and where can we cooperate to move forward. This will not be a long term activity, but a very focused short-term effort which we expect will go on about a year and we’ll have a solid conclusion at the end, and we’ll see where it takes us. We’re very excited about this and we’ll see what the outcomes will be.

I’ll conclude with a couple of notes. Something we’ve been engaging with more recently, is the concerns of countries with smaller programmes. Two-thirds of the countries that generate spent fuel have small nuclear programs, some with only one or two reactors. But each has the responsibility to deal with their nuclear waste. But building a deep geologic repository can be a very expensive and complicated matter. In the US there was an estimate that building a repository will cost about US$50 billion. I don’t think it all will be quite that expensive, but the order of magnitude is probably about consistent around the world. So, if you have one reactor does that mean you have to spend tens of billions of dollars to build a deep geologic repository? Does that make sense? And if it doesn’t make sense what are your alternatives?

All countries must certainly keep moving in the national programme direction. There are some people that choose to continue studying the issue, I think they assume because the Pu-239 has half-life of 24 000 years, if they keep studying it long enough the problem will go away. You can look for ‘take back’ programs, there are some history of that, but it is not as common as it used to be. Third-party repositories have been talked about but never really accomplished. And there’s also the issue of shared repositories. Today, we see more discussion in this area. Clearly, multinational schemes will not be easy to accomplish for many obvious reasons. They’re very politically sensitive issues and there are many legal and regulatory issues that have to be dealt with.

IFNEC, the International Framework for Nuclear Energy Cooperation, did a study on this a few years ago and highlighted that there is an opportunity for cooperation that countries that are interested in shared repositories should approach a dual-track approach, maintaining their current national programmes but exploring multi-national approaches. And at the same time, countries will need to clarify many very important issues, preconditions, national and international laws, safeguards requirements, issues, liabilities, many legal, political issues that have to be settled.

We believe that multi-national cooperation could help national programmes to progress deep geological repository programs. We believe that, all nuclear countries have to have a viable approach to deal with their spent fuel. And multi-national options should not be used as an excuse not to complete the domestic work. But that said, there are obvious benefits particularly to small programmes for shared repositories and we should explore it, we shouldn't be afraid of exploring it. We should continue these explorations, and what we should make sure is we do it in a way that it doesn't slow the progress that many of you are making in your national programmes. So, finding that balance is something that we're working on and will continue working with everyone, including the IAEA and the European Commission.

I will end it there and thank you very much for your attention.

2.4. DIRECTOR EUROPEAN COMMISSION NUCLEAR ENERGY, SAFETY AND ITER'S OPENING REMARKS

Opening speech as provided, verbatim.

Massimo Garribba
Director of DG ENER Directorate D-Nuclear Energy, Safety and ITER

Good morning ladies and gentlemen.

Let me start by saying that in Europe we have at the moment 14 member states that use nuclear energy and 14 that don't. But if I take the lead, from the two previous speakers, we also published a 2050 scenario last November that puts nuclear energy at 15% on average in 2050 as a source of energy in the EU, accompanied by a massive deployment (80–85%) of renewables technology. In such a scenario, the safe and responsible management of radioactive waste is a fundamental element to keep nuclear energy in the energy mix. At the European Commission, we have been consistently working on that for the past ten years, and we have developed a regional safety legal framework that is taken as a reference in the world with spent fuel and radioactive waste management being a key element of this. The directive is still young, it was born in 2011 and implemented in steps in 2013 by becoming national law in the EU Member States and in 2015 by having Member States national programmes to describe how they're going to put in place their policies for the management of spent fuel and radioactive waste. It is particularly important that roles are clear and that political priorities are transformed into actual projects and actions. The directive is based on a key principle, which is that you cannot transfer burdens the following generations. Therefore, while it's true that we can safely manage spent fuel and we can temporarily storage it already today. But this is no excuse in order to sit back and wait for something else to happens.

One of the key elements of the reporting that was made, and the Radioactive Waste Directive, is the notion of inventories, namely how much radioactive wastes and spent fuel do we have in the EU. So, we have something like 50 000 tonnes of spent fuel generated in the past and Member States estimate that this will rise to 80 000 tonnes by 2030. And if you look at this graph, it shows you that the situation is quite diverse. For low and intermediate waste disposal we have a situation that advances and progresses consistently. Whereas for high level waste and spent fuel, very little is happening. This is the situation that we have to face nowadays.

If we look at it in a little bit more detail, the Directive also states that each Member State is ultimately responsible for the disposal of the spent fuel. It is well-known that there are 3 EU member states that are very well advanced in the final disposal techniques. We count on Finland starting operation of the Onkalo repository in 2024, to be followed, let me say in the next decade, to full operation by Sweden and France. But all the other EU member states have projects or programmes or policies that push final disposal solutions back to the period 2050–2130. Therefore, there is a big gap that has to be filled. Indeed, in this kind of situation interim storage is particularly important and needs also to be handled safely.

The next point that comes naturally when you look at the situation, is so how much and how fast is research moving in Europe? And the answer is that only in 4 Member States, I have to add Belgium to the 3 that I mentioned before, there are operational facilities that add to the R&D. There are a number of plans for the 2020s too. They should add to the number of Member States in joining the R&D. Let me say that the situation at the moment is sub-optimal. From the EU side, the Commission operates also in this field through the research programme and it has 2 types of actions. One which is carried out by the Joint Research Centre (JRC) that has a number of sites across Europe. Most of the work on waste is probably concentrated in Karlsruhe in Germany and in Ispra in Italy but all the sites of the JRC collaborate. The second by the Directorate General for Research and Innovation (DG-RTD) that also finances a number of large collaborative projects that bring together different actors and waste agencies in the EU. You can see in the slides that the overall budget over 5 years is around 185 M€ for the 2 actions combined.

To handle nuclear waste, research is one ingredient, but public acceptance and transparency is the other one. The success of both the Finnish and the Swedish operations is rooted in the long-standing collaboration with the populations that live and work near the sites. The Directive recognizes this and puts in law the fact that a number of consultation steps have to be taken with the public in order to define the policy of the national programmes. I think I can say that indeed a transparency policy is one of the most important elements in allowing steady progress for moving ahead in realising spent fuel and high level waste disposal.

We are about to publish a second report on the implementation of the Directive and especially the situation of the inventories, and it is important to recognize that with our Member States we have made a lot of progress in order to ensure the legislation is correctly in place. Some of you will be aware that the Commission open formal dialogues with basically half of the EU member states on the implementation of this Directive, because a) the provisions of transposing it into national law were considered to be insufficient and b) the national programmes that should lead to the waste disposal were considered to be vague and not moving from policy to projects. There are a number of issues that were identified in the first report. One is indeed having a proper system to account for which kind of waste we have and where it is. So quantification of the needs is one of the keys. The second one is to have timeframes which are compatible with the national choices, but they can also be tracked through the use of smart indicators, key performance indicators, that show the progress of the programmes and don't lead to continued postponement.

Through this Directive and collaboration with the IAEA, we have established a system of international peer review through ARTEMIS. Member States, under the Directive, have to ask for a peer review every ten years. The first cycle will have to be concluded by 2023, so we will have a picture of the situation by then. Shared disposal is also mentioned in the national programmes of many of our Member States. Unfortunately, it is mentioned but not detailed. Everybody seems to be willing to share, yet it is not described with whom and under which condition this sharing should happen.

The Directive allows for sharing between EU Member States; however, the Directive doesn't allow for the export of nuclear waste and spent fuel unless there is a working facility in a country outside of the EU which works under the same safety conditions as in the EU, which obviously does not exist today. So, we have for the moment a de facto export ban.

Thank you very much for your attention.

2.5. DIRECTOR GENERAL WORLD NUCLEAR ASSOCIATION'S OPENING REMARKS

Opening speech as provided, verbatim.

Serge Gorlin

Head of Industry Cooperation, World Nuclear Association
on behalf of Agneta Rising, Director General World Nuclear Association

Madame Chair, Deputy Director General Chudakov, fellow panellists, ladies and gentlemen, on behalf of the World Nuclear Association and its Director-General Agneta Rising, I am very grateful for this opportunity to speak to you today about the Sustainability of Used Nuclear Fuel Management. Ms Rising conveys her sincere apologies for her inability to attend today.

The outline of my presentation is as follows. I will begin by giving you an overview of the World Nuclear Association and the Sustainable Used Fuel Management (SUFM) Working Group (WG), before sharing some thoughts on how industry manages used nuclear fuel sustainably. Before offering some conclusions, I will say a few words about the Association's Harmony programme and the important role nuclear energy plays in energy sustainability, as well as a few words on innovation.

Let me begin by giving a brief overview of the World Nuclear Association. Established in 2001, but with roots going back to the 1970s, we are the international organization that promotes nuclear energy and supports the companies that comprise the global nuclear industry. The Association's membership encompasses all aspects of nuclear energy. The Association also runs World Nuclear News, the world's leading online news service on developments related to nuclear power. In addition, we provide administrative support and leadership to the World Nuclear University, a global network committed to training and education of nuclear industry professionals.

The Association's Working Group on Sustainable Used Fuel Management promotes sound, safe, sustainable and proliferation-proof used fuel management. Its mission is to shape industry positions with a view to engaging in the international debate on sustainable management strategies for the back end of the fuel cycle. Addressing the theme of this conference, "learning from the past and enabling the future", one can say that the Working Group's activities of collecting, analysing and distributing leading practice from the past and present and using it to generate recommendations for the future are well aligned.

Perhaps the main message from the past 60 years, is that nuclear energy is an environmentally responsible power generating source that is aligned to the polluter-pays principle. This ensures that nuclear operators make adequate financial provisions to responsibly manage and dispose of radioactive waste and used fuel.

Used fuel management should be conducted in accordance with five defined areas, ensuring the development of a well-structured plan that takes account of forward-thinking technologies coupled with realistic financing models to provide needed protection to the environment and human health, while not inflicting greater burden on future generations.

Upon removal from the reactor core, used fuel embarks on the final stage of its life cycle, with nuclear industry implementing various strategies based on government policy to ensure safe and cost-effective overall management. These strategies are divided into two tracks, the open cycle and the closed cycle.

There is presently a broad consensus among technical experts and policy institutions that the preferred method of ensuring long term safety for high level waste and used fuel is isolation in a deep geological repository. Geological disposal facilities for long lived waste, if properly sited and constructed, will provide passive, multi-barrier isolation of radioactive materials.

Unlike other sources of power generation such as coal and natural gas, used nuclear fuel may be reprocessed or recycled to provide added value as an energy resource. Currently the countries which operate reprocessing facilities are France, India and the Russian Federation. The UK operated reprocessing facilities for light water reactor fuel until recently and will still operate the MAGNOX reprocessing plant until around 2020. China is operating a pilot plant and is looking to deploy an industrial facility. Japan is planning to commission in 2021 its Rokkasho-mura plant. India operates and is developing reprocessing facilities for both thermal and fast reactor spent fuel. And Russia is developing new reprocessing technologies and is increasing its reprocessing capacity.

Used nuclear fuel has been and is successfully transported by truck, rail, and ship using specially designed casks. To date this transport has been to reprocessing plants and to centralized interim storage facilities. The transporting of used fuel is a well proven activity based on meticulous planning. To date it has enjoyed excellent safety record, something that the nuclear industry is determined to maintain.

Until a deep geological repository is operational, used nuclear fuel will have to be placed in interim storage at the reactor site or in a centralized facility. While interim storage is technically feasible, it does raise a concern that the storage of the fuel is not the final solution for it. This is why, echoing the comments of my fellow panellists, a state should proceed with siting, constructing and operating a deep geological repository without unnecessary delay, or they should consider used fuel reprocessing.

If we look at the data from the IAEA, and indeed this is backed up by the sustainable used fuel management working group's own survey in 2017, the start of final disposal is not imminent in most cases. Projects in France, Sweden and Finland are the most advanced countries where engaging and communicating across a wide range of audiences and platforms to involve citizens in developing deep geological disposal projects has taken place. Again, referring to the theme of the conference, there are lessons here from the past that can enable the future.

The accumulation of used fuel is seen by many as a significant reason to oppose nuclear energy, notwithstanding the proven solutions that exist. In this context I'd like to commend the IAEA, OECD-NEA, and European Commission for their collaborative publication 'Status and trends in spent fuel and radioactive waste management', which clearly and concisely explains the status quo with regard to spent fuel management. The World Nuclear Association was proud to be part of this standing committee for this publication. Showing the ability to successfully manage used nuclear fuel will help ensure nuclear energy is able to continue to play an important function to decarbonise our electricity generation and to protect people from the dangers of air pollution. To meet the growing demand for sustainable energy, we will need nuclear to provide at least 25% of the world's electricity by 2050 as part of a clean and reliable low carbon mix. Achieving this means nuclear capacity must triple, globally, by 2050. The Harmony programme is a global initiative of the nuclear industry that provides a framework for action, working with key stakeholders so that barriers to growth can be removed.

While we can claim to have solutions today to manage used fuel, we can never stand still. Striving for continuous improvement is the only guarantee of sustainability. The global nuclear industry is continually innovating to promote enhanced fuel performance along with better management of radioactive waste while improving nuclear safety culture. These advancements achieved today will provide the impetus for tomorrow's enhancements in nuclear energy and radioactive waste management.

There is a natural progression of innovative solutions in the nuclear industry including for used fuel management. These solutions include the development of interim storage solutions, recycling of reprocessed uranium and the development of fast reactors.

In conclusion, it must be recognised that the infrastructures and technologies are available to provide for the efficient and safe management of radioactive waste and used nuclear fuel. While the timeline varies from country to country when a deep geological repository will be sited, constructed and operational, there are adequate interim storage methods available to store used nuclear fuel until such time these facilities become operational. However, one must caution that unnecessary prolonged delays will erode public confidence that used fuel can be satisfactorily managed and potentially undermine nuclear power's role in combatting climate change. Lastly, I would like to add that the global nuclear industry has the competency to mitigate the various risks and uncertainties associated with used fuel management, while it constantly continues with developing implementable innovative solutions to increase the efficiency and safety of the entire nuclear fuel cycle.

Thank you for your attention.

3. SUMMARY OF TECHNICAL SESSIONS AND INVITED PAPERS

3.1. TRACK 1 – NATIONAL STRATEGIES FOR SPENT FUEL MANAGEMENT

Overview prepared by D. Hambley (United Kingdom), **Track Leader**

Several approaches to spent fuel management (SFM) have been and will be adopted around the world due to differences in the implemented technologies and organisational arrangements which, in turn, arise from a range of technical and societal factors. There is a clear consensus that spent fuel management must encompass all activities from discharge from the reactor core to emplacement of fuel and/or waste in a disposal facility.

Embarking countries are consciously addressing back end management before building nuclear power plants and there is evidence of an intent to ensure disposal facility development is progressing alongside the development of power plants to provide comprehensive back end solutions before the end of generation. Such plans are technologically feasible and address intergenerational equity issues faced by many mature nuclear countries.

Some embarking countries are building resilient management strategies by providing alternative options in the events that could delay or foreclose baseline management options, clearly drawing lessons from more mature programmes.

There is growing evidence that a combination of consistent policy, effective public engagement and education, strong regulation and commercially-led delivery provides a sound approach for effective delivery of a comprehensive SFM system. These principles are being adopted in a range of countries.

There is evidence of a trend towards delaying implementation of fast reactor-based closed cycles due to economic conditions arising from ready availability of recoverable natural uranium, although this remains an aim for the future in many countries with a strong recycling programme. Nevertheless, necessary development of fast reactors and associated fuel cycles continues and provides a valuable role in developing the technologies that would support future deployment. In the meantime, development of fuel and fuel cycles that enable multi-recycling in thermal reactors is being pursued by France and the Russian Federation to achieve benefits in resource use and environmental impact beyond the current mono-recycling practice. The trend in these countries and Japan is towards a system that manage Pu inventories across the whole fuel cycle via its reuse rather than to produce Pu for future needs.

A number of national strategies reflect the need to make available sufficient spent fuel (SF) storage capacity to bridge the gap between the generation of spent fuel and the foreseen commissioning and operation of deep geological disposal facilities. The industry continues to develop safe technologies for longer term fuel storage. However, a number of such systems, particularly large capacity canister-based systems, are not compatible with current disposal concepts and clarity on the exit strategies is required to demonstrate effective holistic management.

Summary Bullets

- There is evidence from many countries, especially newcomer countries, of learning lessons from the past and developing robust, integrated strategies for managing SF.
- Recycling of fuel continues to play an important role and there is a new focus on developing multi-recycling in thermal reactors and partitioning in advanced reactors.
- Implementation of closed fuel cycles based on fast reactors is being delayed in many countries, although development of large reactors and associated fuel cycles continues in some countries.
- There is evidence of greater attention being given to impacts of fuel cycle on disposal and vice versa, but there is a need for urgency on work to understand optimization of the whole back end and to actively implement these strategies on the ground.

Session 1.1 – National Strategies for Spent Fuel Management

Session Chairs: D. Hambley (United Kingdom) and T. Saegusa (Japan)

Session 1.1 comprised of five papers, one from the IAEA, one from United Arab Emirates, one from the European Commission, one from Kenya and one from Indonesia.

- **Paper ID#210 by C. Xerri (IAEA)** presented a summary of the status of nuclear power and the IAEA's role in supporting the implementation of Atoms for Peace and sustainability of nuclear power production. He noted the importance of a comprehensive policy framework in all states using nuclear power and the benefits of undertaking periodic reviews. He proposed three future scenarios for nuclear power which depend on societal and economic conditions and, most importantly, covering how the industry innovates and delivers its responsibilities.
- **Paper ID#209 *(Invited)* by H. Alkaabi (United Arab Emirates)** presented the status on nuclear power plant development at Barakah site and the planned nuclear fuel and waste storage arrangements at the site. The framework for planned off-site permanent disposal sites was discussed. The arrangements for managing longer term liabilities for decommissioning and fuel disposition were drawn from international experience and the current state of development, which includes a baseline plan and an expectation that they will be in place prior to the start of generation.
- **Paper ID#194 *(Invited)* by M. Martín Ramos (European Commission)** presented an overview of the purpose and principle for nuclear related activities of the EU Joint Research Centre with particular emphasis on the research and training programme, the Horizon 2020 research programme and their interactions. Recent work on research to understand the effects of long term storage on irradiated fuel, accident conditions relevant to transport and the behaviour of spent fuel in disposal systems were highlighted.
- **Paper ID#23 by H. Mpakany (Kenya)** presented the policy, strategy frameworks for nuclear and radioactive activities and the action plan for development of four nuclear power plants in the 2030s. The existing framework for radiation protection, included interfaces with environmental and maritime legislation. The new policy framework for introduction of nuclear power addresses all aspects of back end management and supporting technical and societal needs. The selected fuel cycle option is an open cycle and the importance of establishing a final repository simultaneously was also recognized.
- **Paper ID#20 by S. Prihastuti (Indonesia)** presented arrangements for management of spent fuel from existing research reactors in Indonesia. For research reactor (RR) fuel there is regular fuel condition monitoring and regulatory reporting and the end point for the fuel is return of fuel to the supplier country, USA. There is a plan to construct a nuclear power plant in the future and initial steps in updating the legal framework are described.

Session 1.2 – National Strategies for Spent Fuel Management

Session Chairs: M. Martín Ramos (European Commission) and S. Prihastuti (Indonesia)

Session 1.2 comprised of five presentations, one from Brazil, one from Belarus, one from Slovakia, one from Spain and one from France, plus a summary overview of the poster session by the Conference Chairwoman (USA).

- **Paper ID#4 by A. Vidal Soares (Brazil)** presented the need for managing spent fuel in Brazil. Mr. Vidal Soares provided an overview of the electricity generation mix, to highlight the role of the 2 nuclear power reactors in Brazil, as well as the intention and ongoing work to extend the operation of both units by 20 years. It is necessary to increase the spent fuel interim storage capacity from Unit 1 and Unit 2 of Angra, as the spent fuel pools are almost full. The presentation provided some highlights of the implementation of the independent spent fuel storage facility, based on dry cask technology including data on the number of casks, loading campaign magnitude and calendar, as well as other technical, regulatory and economic information.
- **Paper ID#80 by A. Kuzmin (Belarus)** presented a comprehensive analysis of the spent fuel panorama in Belarus. Mr. Kuzmin introduced the legal and regulatory framework, including the international treaties and conventions to which Belarus is a member party. Based on the spent fuel generation prospective in the country with the commissioning of the Ostrovets nuclear power plant and a comparison between the available spent fuel management options (direct disposal or reprocessing), considering the developments in the country that supplies the technology, preferred option for Belarus is the shipment of spent fuel to the Russian Federation for reprocessing and the return of the corresponding waste. The strategy encompasses all the necessary steps (facilities, legislation, human and technical resources, etc.) aiming at practically implementing the preferred strategy.
- **Paper ID#143 by J. Vaclav (Slovakia)** presented the extension of the spent fuel storage capacity of Jaslovské Bohunice. Mr. Vaclav explained the context and rationale for the first extension of the wet storage facility (shipping the spent fuel to the Soviet Union for reprocessing was no longer possible after the late ‘80s) and the current need to further extend the capacity, this time with a dry casks storage system to enable the accommodation of the fuel generated in the Mohovce nuclear power plant. The presentation provided details on the safety enhancements requested by the regulatory body to the operator, based on the findings and analysis of the safety assessments, with some highlights on postulated accidents and seismic requirements.
- **Paper ID#101** ***(Invited)*** **by F. Lentijo Robledo (Spain)** presented the Spanish national policy and strategy for spent nuclear fuel, high level and special radioactive waste management. The presentation covered topics such as the principles and outline of the national policy reflected in the General Radioactive Waste Plan; the national framework, including the laws, the entities – operators, Ministry, Regulatory Authority, Radioactive Waste Management Agency – involved, their tasks and responsibilities; the current inventory and current spent fuel management situation (which includes cool-down in the nuclear power plants spent fuel pools, storage in dry casks systems built at the different sites); an overview of the implementation of the planned centralized dry storage facility (with information on the site selection process, the design, the operation, the safety requirements and the licensing process), as well as the future actions.

- **Paper ID#204** ***(Invited)*** **by E. Touron (France)** presented the future perspectives for the fuel cycle in the framework of the French Strategy for Energy, which considers a reduction of the share of nuclear energy in the electricity mix, maintaining reprocessing and recycling of nuclear fuel. In the short term, mono-recycling in the current fleet of reactors is maintained, while preparing for the use of MOX in the larger reactors. In the mid-term, the aim is to develop multi-recycling with a view of closing the cycle in the longer term, with Generation IV fast neutron reactors. The presentation of Mr. Touron listed the most outstanding R&D challenges for the proposed developments, and highlighted the benefits, in terms of radioactive waste reduction and natural uranium resource savings, and the different steps to be taken towards the closure of the fuel cycle.
- **Paper ID#203 by S. Y. Pickering (USA)** and Chairwoman of the Conference provided an overview of the posters displayed during the Conference. The overview consisted of a representative selection of posters to illustrate the six tracks in which the Conference is structured. Track 1 encompasses posters on national strategies of each countries. Track 2 encompasses posters on the behavior of spent fuel under long term wet and dry storage, ageing management, safety, operation and economics. Track 3 encompasses posters on regulatory requirements, for all normal, abnormal and accident conditions of transporting spent fuel. Track 4 addresses recycling of fuels using different technologies. Track 5 includes, for example, analysis of the impact of advanced separation, partitioning and transmutation, or the use of the Thorium cycle and its impact on the deep geological disposal of spent fuel. Posters in Track 6 address topics such as safeguards, knowledge management and technical considerations for deep geological repositories. Track 7 encompasses posters on sustainable development, spent fuel management and considerations to enhance public understanding. Ms. Pickering also invited the participants to visit the e-posters, which consist in a brief presentation (around 10 minutes) and give the opportunity to have discussion between authors and audience.

Session 1.3 – National Strategies for Spent Fuel Management

Session Chairs: S. Salzstein (USA) and E. Touron (France)

Session 1.3 comprised of six papers, one from India, one from China, one from USA, one from Japan, one from United Kingdom and one from Russian Federation.

- **Paper ID#74** ***(Invited)*** **by J. Yadav (India)** presented the nuclear power development strategy in India and the current and planned nuclear power plants, spent fuel storage facilities and reprocessing facilities. The objectives of current advanced recycling research were described, which included actinide separation and industrial isotope separation. The commercial viability of isotope extraction was still uncertain but the off-setting of disposal costs and environmental impact would need to be factors considered in the assessment.
- **Paper ID#171** ***(Invited)*** **by Y. Guoan (China)** presented a summary of current nuclear power generation, its planned developments and spent fuel accumulation expected by 2050. He described the research facilities that are providing the basis for near term deployment of a closed fuel cycle using fast reactor technology, the roadmap for implementation and progress on the construction of a commercial demonstration fast reactor and commercial reprocessing and MOX fabrication facilities. Planned transportation systems for transfer of spent fuel to centralized storage facility were

described alongside the deep geological repository (DGR) development programme. Research on accelerator driven fast reactors and molten salt reactor concepts were outlined.

- **Paper ID#180** ***(Invited)*** **by P. Lyons (USA)** presented a summary and comparison of the approaches to and delivery of two disposal programmes: the Waste Isolation Pilot Plant (WIPP) for defense transuranic waste and Yucca Mountain for commercial light water reactor (LWR) spent fuel. Particular focus was placed on the societal and political processes associated with the developments and the lessons they provide for future strategic nuclear facility siting process.
- **Paper ID#58** ***(Invited)*** **by K. Yoshimura (Japan)** presented the status of Japan's policy on spent fuel management and its new principle of stabilizing separated Pu in the fuel cycle. The current status of storage and reprocessing and related facility construction programmes was given and changes to the arrangements for funding reprocessing operations were described. Japan's recently defined strategic roadmap for fast reactor development was presented. The factors leading to Japan's revision of its geological disposal legislation was summarized and latest developments in the new site selection process were summarized.
- **Paper ID#160** ***(Invited)*** **by P. Hallington (United Kingdom)** presented a summary of the United Kingdom's oxide fuel recycling business, summarizing its operational history and timeline. He highlighted the high standards achieved in design and operations which he ascribed to the commercial framework that was maintained throughout the facility's operation. The Thermal Oxide Reprocessing Plant (THORP) has been run as a commercial business from start to finish and that finish was a direct result of the reduction in both domestic and international reprocessing requirements.
- **Paper ID#25** ***(Invited)*** **by A. Khaperskaya (Russian Federation)** presented a summary of the recent and current developments in spent fuel management, with particular emphasis on the developments at centralized facilities at the Mining and Chemical Combine (MCC). The opportunities provided by innovations in thermal fuel recycling were described, opening a path for increasing the benefits from recycling operations. Longer term plans, particularly in relation to development of fast reactor systems, were also summarized. During the conference, supplementary details were given by other presenters from the Russian Federation.

Paper ID#209
UAE PROGRESS ON THE DEVELOPMENT OF A NATIONAL STRATEGY ON THE MANAGEMENT OF NUCLEAR WASTE

HE H. ALKAABI
United Arab Emirates

Transcription from talk, as verbatim

Thank you very much for this opportunity.

My plan is to give you a little bit of an overview of the latest development of the UAE nuclear power plant project, but also, relevant progress in relation to the waste management and the spent fuel management in terms of policy and also legislation.

A background of the UAE nuclear programme. In 2008 the government issued its nuclear policy on the evaluation and potential implementation of nuclear power. The decision to establish the UAE nuclear power programme was matured in 2009 by taking concrete steps and the establishment of the nuclear law, and then establishing the institutions relevant to the nuclear power. In late 2009, the decision to contract for the first four nuclear power reactors in the UAE took place. Today the progress, as you see in the pictures, we have four nuclear power reactors under advanced stage of construction. Unit one is already completed, the commissioning process is underway .

With regard to the licensing process, the first application for the operational license for first two units has been submitted in 2015. The UAE nuclear regulator has been reviewing these applications since then. Multiple other licenses were given in terms of the transport of fresh fuel, which is now on site. Currently the progress is to the final phase of the evaluation of the operation readiness of unit 1. The updated schedule for Barakah commissioning is early next year, 2020, for the fuel load and that's where the current progress is aiming.

In relation to spent fuel and waste management, they are already existing plans in Barakah nuclear power plant and progress in terms of relevant policies. It is important to note that Barakah project has took many lessons learned from industries and previous practices. The storage pool for spent fuel has been increased in the design of Barakah nuclear power plant to take up to 20 years of spent fuel in the wet pools. Also, there is currently an initial planning for establishing, what is referred to as the independent spent fuel storage installation, which is basically a dry storage, that will be ready by the time or before the time spent fuel is moved from the pools to the dry storage. Also, of note, Barakah nuclear power plant site has 10 years storage capacity of low and intermediate level radioactive waste already planned in the design. The Decommissioning Trust Fund (DTF) is yet to be established, but a lot of work has gone into it and expect this to be finalised in the near future , we'll talk a little more about it in later slide.

In terms of the legislative framework that governs the spent fuel management in the UAE, I think it's important to refer to the nuclear law of 2009. Which is basically translated the policy commitment of 2008 into a binding law. The nuclear law established FANR, which is the Federal Authority for Nuclear Regulation, as the Authority responsible for safety, security, safeguards and radioactive protection in the nuclear sector to give FANR a specific responsibility but also the law has very specific provisions in relation to spent fuel management.

Just to mention a few, the law requires the licensee to be responsible for the safety, predisposal management including safety all the way to the delivery of the fuel to a designated entity, which is known internally as the waste management organisation, an entity that is yet to be established. The nuclear law also talks about the establishment of the decommissioning fund, which is an important milestone for our waste management strategy within the UAE. The nuclear law also talks about the final role as opposed to other stakeholders in terms of the development and requirements in developing responsibilities among the stakeholders as we go forward.

In terms of the current work and relation to establishment of waste management organisation, we have been in the last few years looking at different practices internationally and the practice of other countries in relation to establishment of the waste management organisation. We saw many examples and different concepts. Some are more government focused. Some are more industry focused. And the current thinking now in the UAE it will be, again, mostly focused on the management by the industry but the government will retain certain role in terms of defining or approving the strategy, technology, the fees, and so on. That role is to be established or to be finalised as we issue the decommissioning fund and the establishment of the policy which I will mention a little bit later in the slide.

In terms of regulatory framework, we already have some elements established as a part of multiple regulation that has been issued since 2009, that address some aspects related to the spent fuel management, including radiation dose limit, and issues and regulations related to decommissioning of the facilities. We have two upcoming regulations that are currently in draft mode. Draft mode means under consultation by different stakeholders in the UAE. One is related to the Decommissioning Trust Fund, which when finalised will implement the decision of the cabinet and the other one is related to the near surface radioactive waste disposal facility regulation.

As I mentioned in terms of the current planned activities, ENEC, which is Emirates Nuclear Energy Corporation, the owner of the nuclear power plant, has already plans and designs in the plant, a dry storage facility within the site itself. Feasibility study has been conducted to identify the ideal location within the current site boundary. They have followed the strategy and an approach to minimise the volume of waste generated in Barakah. The Barakah nuclear power plant, will also have a 10 years storage capacity when it comes to low-level and intermediate-level radioactive waste, that's a part of the plans already at Barakah. The long-term storage disposal through construction of a separate low and intermediate-level radioactive waste storage disposal facility near Barakah, near power plant, that's already under planning.

The Decommissioning Trust Fund is referenced in the nuclear law, the provision that says FANR will make recommendation to the cabinet of ministers to establish a decommissioning fund. The decommissioning fund will basically be a fund that will collect fees to be invested for future decommissioning activities for Barakah nuclear power plant. This fund now has not been established yet, but a lot of work has gone into it in terms of drafting, developing relevant regulation around it and so on. The idea is to have the decommissioning fund be established and operating before the actual operation of the Barakah nuclear power plant. So, the fees will be collected before the actual first megawatt that's been generated by Barakah. The decommissioning fund took a lot of lessons learned or benefited from experiences internationally in terms of what other countries has been doing in this process, but it's also part of the UAE mission and commitments in the policy. To have a successful and sustainable nuclear power plant programme, we have to ensure that responsibility for the future is taken on early in programme including the establishment of a decommissioning fund but also clear policies when it comes to the management of spent fuel and radioactive waste.

The cost estimate, of course, has been one of the big points in terms of discussion between different stakeholders. We have taken a reference scenario where UAE in 90 years will establish a geological repository. We have some references that was used as a benchmark, and as well available data internationally to calculate this cost based on a scenario where this site will be operated in 90 years. we also took into consideration the decommissioning cost from various studies available including the OECD-NEA or other publicly available information in terms of cost of decommissioning. The graph shows the timeline for decisions, or required actual spending related to the decommissioning fund at different timelines of the upcoming 100+ years.

So, in terms of the next steps, one of the things, I see this as a package to going forward, is the adoption of the decommissioning fund in its final form will be accompanied with the establishment of long term government policy or announcement of such policy for the management of spent fuel waste and radioactive waste . In the current policy, we have already developed elements related to waste management including potential timeline, including establishment of responsibilities, division when it comes to implementation of the strategy related to the waste management. Actual practical measures to implement the policy in terms of who would be developing the strategy, who would be approving the strategy, who would be updating the strategy, and I'm talking about decisions such as the technology selections, timelines and issues related to how much and where the fund will be and so on. This policy is planned, again, to be issued at the same time when the decommissioning trust fund will be announced or established before hopefully the operation of the first unit in the UAE. Work in relation to the low-level waste disposal within the site in Barakah, is already ongoing. There's also a planned work in terms of initial studies in relation to geological repository. And I would like to mention here that the policy draft and its current form takes into consideration a base scenario, which is that the UAE will establish a geological repository and will start conducting some desktop studies and go forward to further studies and work with our international partners in terms of building on the experience that is developed internationally. The policy scenarios will include different decision points at different times. Initially we had options such as fuel lease, an option to return fuel to other countries for potential reprocessing and bringing back vitrified waste to the UAE. Some of these options are no longer valid, since the UAE, once it starts operating, will start producing spent fuel and would require more practical policies. however, issues such as technology selections and other related elements can wait a little , but the idea is to strike a balance in terms of decisions related to technology and those decision related to required actions , taking into considering of course developments when it comes to technology. And this is the reason why the government has decided to not to keep the question open but have a base scenario instead . As we go on with the implementation of this base scenario, there will further decision points that could change based on additional information available to the UAE at the time , or additional opportunities available regionally, internationally and so on.

Lessons learned that we as a newcomer learned from other or from experiences we gathered from other nuclear power programmes, included of course planning early! That's one of the most important elements I think. This is the reason why the UAE today has started thinking about this so early in the programme started first in policy in 2008 and as of now we still, before operating, we have a lot of elements of this strategy in place. Of course, the other consideration is picking a site, thinking about the locations, thinking about strategy in terms of "are you going to locate everything within the site?" and so on. planning early also allows to have a better general understanding in terms of how much your programme will accumulate in terms of waste generated in the operational phase of the programme but then the last point, of course, making sure you have money for it when it's needed for decommissioning. That's the thinking or the work established in the UAE related to decommissioning fund so far, it really goes to this point why we have to start early with this process. Of course, getting assistance from international partners, including the IAEA and other experienced countries, is a key for us as a newcomer country. We're building these cooperation frameworks, we're using a lot of existing frameworks and we continue to be interested and work with our international partners.

With this I thank you and I will be ready to answer any specific questions you may have.

Paper ID#194
EUROPEAN COMMISSION'S JOINT RESEARCH CENTRE RESEARCH ON THE SAFETY OF SPENT FUEL AND HIGH LEVEL RADIOACTIVE WASTE MANAGEMENT

M. MARTÍN RAMOS
European Commission's Joint Research Centre
Brussels, Belgium
Email: Manuel.MARTÍN-RAMOS@ec.europa.eu

V.V. RONDINELLA, T. WISS, D. PAPAIOANNOU, R. NASYROW
European Commission's Joint Research Centre
Karlsruhe, Germany

Abstract

The management of the spent fuel in the EU is addressed in alignment with Council Directive 2011/70/Euratom, which aims at the safe and responsible management of radioactive waste and spent fuel in order to avoid imposing undue burdens to the future generations; at ensuring the highest levels of safety; and at ensuring transparency and the involvement of the public in the decision-making process. Twenty-one EU Member States manage about 59 000 tHM of spent fuel generated in past and current nuclear power generation and nuclear research activities. Each year, about 3200 tHM of additional spent fuel are generated. Some Member States reprocess spent fuel and some others have decided to keep this option open. The majority of the EU Member States have opted for direct disposal of their spent fuel. Right now, the EU does not have in its territory any facility for the disposal of spent fuel, high level and long-lived radioactive waste. Finland, Sweden and France expect to start the operation of their deep geological disposal facilities within the next two decades, while the rest of the Member States with nuclear programmes have planned operating disposal facilities in the time interval 2040–2130, with a peak in the decade of 2060–2070. Long term or extended interim storage is thus instrumental in the national strategies for the management of spent fuel prior to reprocessing or disposal. The Euratom Research and Training Programme contributes, within its portfolio of activities, to the safe management of spent fuel and radioactive waste. This is done through indirect research and innovation activities to which the European Union provides financial support, and which are undertaken by EU Member States research entities, and through direct research and innovation activities undertaken by the Commission through its Joint Research Centre (the 'JRC': the European Commission's science and knowledge service). This paper provides an overview of the JRC areas of research relevant for safety of spent fuel (and high level radioactive waste), which cover all stages of spent fuel management since it is removed from the reactor: cooling in the spent fuel pool; handling, transport, storage (with particular emphasis on long term storage); retrieval, handling and transportation after storage; disposal in a deep geological formation, and long term safety aspects thereafter. The paper highlights the main achievements, and the main challenges, stressing the relevance of the experimental work carried out on "real" spent fuel in JRC's research infrastructure, which include hot cells and other shielded facilities that are relatively rare or even unique.

1. INTRODUCTION

It is up to each European Union (EU) Member State to choose whether or not to use nuclear power in its energy mix. Fourteen Member States have nuclear power plants currently operating, which generate around one fourth of the electricity in the European Union. Overall, twenty-one EU Member States manage about 59 000 tHM of spent fuel generated in past and current nuclear power generation and nuclear research activities, and about 3200 tHM of additional spent fuel are generated each year. Pursuant to Council Directive 2011/70/EURATOM of 19 July 2011 establishing a Community framework for the responsible and safe management of spent fuel and radioactive waste [1], EU Member States shall, among other obligations, establish and maintain national policies on spent fuel and radioactive waste management ensuring a high level of safety to protect workers and the general public against the dangers arising from ionising radiation.

The management strategy for spent fuel from nuclear power plants and research reactors in the few EU Member States that consider the spent fuel as a valuable resource consists of reprocessing and reusing the fissile and fertile material recovered and disposing of the high level radioactive waste resulting from the process. Most Member States consider spent fuel as radioactive waste, and thus have opted for its direct disposal. A few of the

Member States that have adopted and follow the direct disposal policy keep open the option of reprocessing of spent fuel, and plan to take the final decision in the future.

Regarding spent fuel from research reactors, a few Member States opt for returning it back to the countries in which it was manufactured, and a small number of Member States with training and demonstration reactors have not yet defined the strategy for the long term management of their spent fuel.

There is a general consensus at technical level that the safest and most sustainable option for the management of high level radioactive waste and spent fuel (when considered as waste) is its disposal in a deep, stable, geological formation. There is not yet any deep geological facility in operation for the disposal of high level radioactive waste or spent fuel in the European Union, nor in the rest of the world. Finland, Sweden and France expect to start the operation of their deep geological disposal facilities within the next two decades, while the rest of the Member States with nuclear programmes have planned starting and operating disposal facilities in the period 2040–2130, with a peak in the decade of 2060–2070. Taking into account the very long timeframes until disposal facilities are ready to receive high level waste or spent fuel, long term interim storage becomes instrumental in the national strategies for the management of spent fuel prior to reprocessing or disposal.

In effect, to bridge the time gap up to the availability of disposal options, a majority of EU Member States that has or has had nuclear power plants in operation has made or is making available increased storage capacity for spent fuel and high level radioactive waste. Under the current situation, spent fuel will need to be stored under the highest levels of safety for periods of time many decades longer than initially foreseen (and licensed) when the first interim storage facilities were commissioned, spanning up to more than 100 years. It is then crucial to identify and understand mechanisms that may affect the evolution of the spent fuel 'system' (including spent fuel rods and assemblies, structural materials and containers) during long term storage, and to ensure that it will still retain sufficient properties and conditions to stand handling and transportation to the disposal facility, or otherwise take the appropriate measures.

The present paper provides an overview of the EU research, in particular the direct research and innovation activities undertaken by the European Commission through its Joint Research Centre (the European Commission's science and knowledge service) in the area of spent fuel and high level waste safety related to long term storage and disposal in deep geological formations.

2. THE EURATOM RESEARCH AND TRAINING PROGRAMME

The Euratom Treaty establishes that the Commission is responsible for promoting and facilitating nuclear research in the Member States and for complementing it by carrying out a Community research and training programme. These programmes are proposed by the European Commission and are discussed and adopted by unanimous vote in the Council. The programmes are funded by the budget of the Community.

The Euratom Research and Training Programme 2014–2018 [2] and its extension 2019–2020 [3] (the Euratom Programme) is implemented through so called Indirect and Direct Actions. Indirect Actions are research activities carried out by consortia of research institutions from EU Member States and associated countries partially funded by the research budget of the European Union. Direct Actions are research activities carried out by the Commission's Joint Research Centre (JRC). The overall objective of the current Programme is "to pursue nuclear research and training activities with an emphasis on the continuous improvement of nuclear safety, security and radiation protection, in particular to potentially contribute to the long term decarbonisation of the energy system in a safe, efficient and secure way."

The Programme also sets specific objectives for both Indirect and Direct Actions. Specific objectives of the Indirect Actions encompass supporting the safety of nuclear systems; contributing to the development of safe, longer-term solutions for the management of ultimate nuclear waste, including final geological disposal as well as partitioning and transmutation; supporting the development and sustainability of nuclear expertise and excellence in the Union; supporting radiation protection and the development of medical applications of radiation, including, inter alia, the secure and safe supply and use of radioisotopes; moving towards demonstrating the feasibility of fusion as a power source by exploiting existing and future fusion facilities; laying the foundations for future fusion power plants by developing materials, technologies and conceptual design; and promoting innovation and industrial competitiveness; (h) ensuring the availability and use of research infrastructures of pan-European relevance.

Direct Actions constitute an important part of the Euratom Programme and pursue specific objectives: improving nuclear safety, (including nuclear reactor and fuel safety), waste management (including final geological disposal as well as partitioning and transmutation); decommissioning, and emergency preparedness; improving nuclear security, including: nuclear safeguards, non-proliferation, combating illicit trafficking, and nuclear forensics; increasing excellence in the nuclear science base for standardisation; fostering knowledge management, education and training; and supporting the policy of the Union on nuclear safety and security.

The Programme is an integral part of Horizon 2020, the EU Framework Programme for Research and Innovation.

The Commission's proposal for the next Euratom Research and Training Programme 2021–2025 [4], which is currently being discussed at the Council aims at focusing in the same key research areas as the current programme, i.e. nuclear safety, security, radioactive waste and spent fuel management, radiation protection and fusion energy. At the same time, the programme intends to expand research into non-power applications of ionising radiation, and make improvements in the areas of education, training and access to research infrastructure.

With the aim of exploiting synergies and better streamlining both the Indirect and Direct Actions, the new programme aims at a single set of common objectives. Two general ones: to pursue nuclear research and training activities to support continuous improvement of nuclear safety, security and radiation protection; and to potentially contribute to the long term decarbonisation of the energy system in a safe, efficient and secure way.

And four specific objectives: improve the safe and secure use of nuclear energy and non-power applications of ionizing radiation, including nuclear safety, security, safeguards, radiation protection, safe spent fuel and radioactive waste management and decommissioning; maintain and further develop expertise and competence in the Community; foster the development of fusion energy and contribute to the implementation of the fusion roadmap; and support the policy of the Community on nuclear safety, safeguards and security.

The Programme will be an integral part of Horizon Europe, the next EU Framework Programme for Research and Innovation.

3. EUROPEAN COMMISSION'S JOINT RESEARCH CENTRE

The JRC is the European Commission's science and knowledge service. It employs scientists to carry out research in order to provide independent scientific advice and support to EU policy in areas such as agriculture, food security, environment, climate change, innovation, growth, as well as in nuclear safety and security.

The JRC creates, manages and makes sense of knowledge and anticipates emerging issues that need to be addressed at EU level. It develops innovative tools and makes them available to policy-makers. It explores new and emerging areas of science and hosts specialist laboratories and unique research facilities. Its scientific results are highly ranked by international peer systems.

Established as a Joint Nuclear Research Centre by Article 8 of the Euratom Treaty, the JRC draws on 60 years of scientific experience and continually builds its expertise, sharing know-how with EU countries, the scientific community and international partners. With time, the JRC has broadened its field of research to non-nuclear disciplines, which now cover around 75% of its entire activities. It works together with over a thousand organisations worldwide in more than 150 networks whose scientists have access to JRC facilities through various collaboration agreements.

The JRC is funded by the EU's framework programme for research and innovation: Horizon 2020, and by the EURATOM Research and Training Programme for its work in the nuclear field.

The JRC is organised in two Directorates with corporate responsibilities for strategy, work programme coordination and resources, and eight scientific Directorates: six of them deal with Growth and Innovation; Energy, Transport and Climate; Sustainable Resources; Space, Security and Migration; Health, Consumers and Reference Materials; and Nuclear Safety and Security; two are cross-JRC directorates, for Knowledge Management and Competences. The JRC Directorates are spread across six sites in five different countries within the EU: Brussels and Geel in Belgium, Petten in The Netherlands, Karlsruhe in Germany, Ispra in Italy, and Seville in Spain.

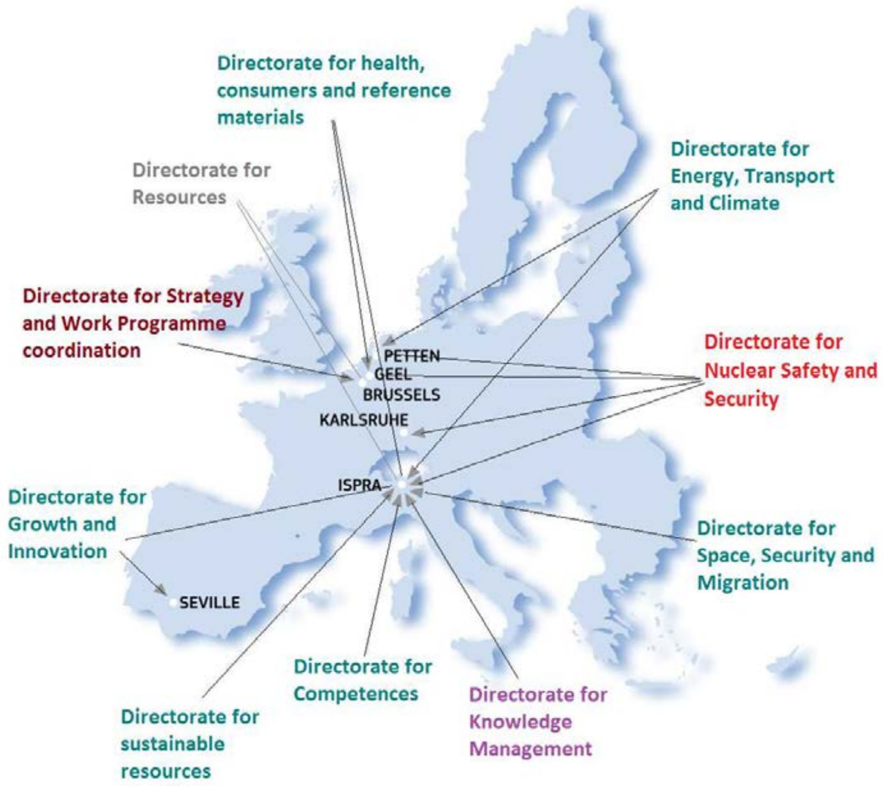

FIG. 1. European Commission's Joint Research Centre sites.

3.1. JRC research and training in nuclear safety and security.

The Directorate for Nuclear Safety and Security employs about 460 scientists, technicians and administrative staff in Petten, Karlsruhe, Geel and Ispra.

The JRC work programme for nuclear activities is structured in about 20 projects on nuclear safety, waste management, decommissioning and emergency preparedness, nuclear security, safeguards and non-proliferation, reference standards, nuclear science and non-energy applications; and education, training and knowledge management. To align with and complement the research and training needs of the Member States, JRC is continuously interacting with the main research and scientific institutions in the EU, and actively participating in technological platforms and associations. JRC also participates as member of the consortium in several Indirect Actions; this allows JRC scientists to engage in top level scientific research together with relevant actors from the Member States, maintaining and further developing scientific excellence. At the same time, the members of the consortia can have access to unique research infrastructure.

JRC's most relevant activities in the nuclear reactor safety research domain encompass, without being exhaustive: advanced mechanical testing methods to address creep fatigue or stress corrosion cracking at high temperatures in corrosive environments (e.g. supercritical water and liquid metals); severe accident modelling and analysis using computer codes (e.g. the European software system ASTEC). The JRC operates the EU Clearinghouse on Operating Experience Feedback, a regional network constituted by nuclear safety regulatory authorities and their technical support organisations that aims at enhancing nuclear safety through further use of lessons learned from Operating Experience. Another key activity is the development, operation and maintenance of EURDEP, EU systems for almost real-time monitoring of radioactivity in the environment, and support to ECURIE, the EU early notification and information exchange system for radiological emergencies.

JRC also carries out research on safety of the nuclear fuel cycle: the scope of these research activities encompasses in-core irradiation behaviour, spent fuel handling, transportation, storage and disposal, and covers normal, off-normal and (severe) accident conditions. JRC developed and further improves and maintains the TRANSURANUS computer code, which is an independent computer code for fuel performance analysis employed by an extensive network of users in the EU and in third countries. JRC research is not limited to current light water reactor (LWR) nuclear fuels but includes also advanced and innovative designs for evolutionary or next generation systems. In particular, JRC investigates safety and safeguards aspects of Generation IV reactors and fuels and is the Euratom implementing agent of the Generation IV International Forum.

In the area of radioactive waste management, JRC R&D activities cover spent fuel and high level waste safety aspects (see chapter 4), and also management of waste from decommissioning and site remediation applications. The projects covering the latter focus on: non-destructive analysis for the characterisation of waste packages; standardisation of free release measurements; development of novel techniques for detection and mapping of contamination; damaged fuel and debris characterization and removal from high activity environments; remediation applications, e.g. tools to analyse in-situ ‘hard to measure’ nuclides, etc.

JRC activities in the field of nuclear security and safeguards focus on four main areas: effective and efficient safeguards (through research on nuclear material detection, characterization, containment and surveillance, and through process monitoring including on-site laboratories); verification of absence of undeclared activities (e.g. through trace and particle analysis, and development of in-field deployable tools); nuclear non-proliferation (e.g. through export control and trade analysis studies); combating illicit trafficking (e.g. through nuclear forensics, equipment development, testing and validation, preparedness plans).

In the standardisation domain, the JRC is a reference entity for reference measurements and data, basic and pre-normative research, and inter-laboratory comparisons. The JRC develops and manufactures standards and reference materials. It is a major European provider of nuclear data and standards for nuclear energy applications, due also to its unique scientific infrastructure. The main repositories for these data are the databases of Nuclear Data bank of the NEA-OECD and the IAEA, which provide open access to the data.

JRC has relevant research activities in the field of nuclear science applications, such as accelerator-based nuclear measurements, basic properties of actinides, and radionuclides for special applications, including nuclear medicine and space applications.

JRC activities in knowledge management, education and training include organisation and active participation in expert and scientific conferences, and the organization and implementation of education and training initiatives such as the European Nuclear Security Training Centre (EUSECTRA), European Safeguards Research and Development Association (ESARDA), education and training of Euratom and IAEA nuclear inspectors, European Learning Initiatives in Nuclear Decommissioning and Environmental Remediation (ELINDER), international schools and courses on radioactive waste management and decommissioning, nuclear safety, security, nuclear data, etc. Students and young researchers can access JRC nuclear research facilities through several programmes enabling them to perform research projects as part of their academic or post-academic curricula. Access to JRC nuclear research infrastructure is an area that will be further expanded and enriched during the next Framework Programme.

3.2. JRC nuclear research infrastructure.

The nuclear research experimental facilities of the JRC are distributed among the sites of Geel (Belgium), Petten (the Netherlands), Karlsruhe (Germany) and Ispra (Italy).

JRC-Geel research infrastructure mainly focuses on nuclear data, radioactivity metrology, and nuclear reference materials:

- The neutron time-of-flight linear accelerator (GELINA) is a pulse white spectrum neutron source with the best time resolution in the world. GELINA combines four specially designed and distinct units: a high-power pulsed linear electron accelerator, a post-accelerating beam compression magnet system, a mercury-cooled uranium target, and very long (up to 400 m) flight paths;
- The Tandem accelerator based monoenergetic fast neutron source (MONNET) is a vertical 3.5 MV Van de Graaff accelerator that produces continuous or pulsed ion beams, providing a stable neutron field for more than a week. The combination of both facilities GELINA and MONNET makes JRC-Geel one of the few laboratories in the world capable of producing the required accuracy for neutron data needed for the safety assessments of present-day and innovative nuclear energy systems;
- The radionuclide metrology laboratories consist of a cluster of instruments for high precision radioactivity measurements (RADMET laboratories) and the high activity disposal experimental site (HADES): a laboratory for ultra-sensitive radioactivity measurements 225 m deep underground;

— Nuclear reference materials laboratories for the preparation and provision of certified nuclear reference materials and reference measurements (METRO), and well-defined and well-characterised samples for nuclear data measurements (TARGET). These laboratories encompass equipment for mass spectrometry, chemical sample preparation in glove boxes, substitution weighing in glove boxes, robot systems, and production of reference particles and UF_6 reference measurements.

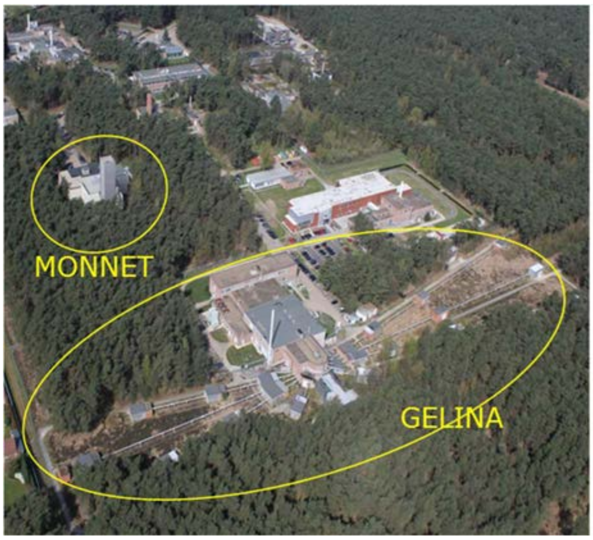

FIG. 2. Accelerators for nuclear data measurements in JRC-Geel.

JRC-Petten hosts and operates laboratories for the assessment of materials and components performance under thermo-mechanical loading, corrosion, and neutron irradiation:

— The high flux reactor (HFR, owned by the EC-JRC but operated by the Dutch NRG) is one of the most powerful (45 MW) multi-purpose materials testing research reactors in the world. The tank in pool type light water-cooled and moderated reactor provides irradiation facilities and possibilities in the reactor core, reflector region and in the poolside facility, as well as neutron beams;
— The laboratory for the ageing of materials in LWR environments (AMALIA) is a laboratory for aqueous corrosion and stress corrosion cracking investigations, a unique facility encompassing four recirculating water loops with 6 autoclave systems, all featuring full water chemistry control. The autoclaves (T_{max} = 650°C, P_{max} = 360 bar) are equipped with environmental mechanical testing facilities (slow strain-rate tensile tests, crack initiation and crack growth rate tests, fracture mechanics, cone-mandrel tests, small-punch tests), electrochemistry, electric impedance, DC potential drop, and acoustic emission monitoring, to assess coolant compatibility and materials degradation issues in light water reactor environments;
— The Structural Materials Performance Assessment laboratories (SMPA) are used for the mechanical performance characterisation, life assessment and qualification of structural materials for present and next generation nuclear systems. The test installations include 3 servo-hydraulic and 3 electro-mechanical universal test machines for (thermo-)mechanical tests, low-cycle fatigue, and fracture mechanics tests, 11 uniaxial creep rigs, 5 small-punch creep rigs, 2 Charpy test rigs, a dedicated test rig for thermal fatigue tests of tubular components, and a nano-indentation hardness tester (–150°C to + 700°C). Depending on the application, temperature control ranges from cryogenic (liquid nitrogen) to high temperatures (induction heaters, radiation heaters and resistance furnaces);
— The Microstructural Analysis Infrastructure Sharing laboratory (MAIS) is a user lab for microstructural characterisation and materials degradation studies. The facilities include scanning electron microscopy, transmission electron microscopy and atomic force microscopy (AFM), optical microscopy, metallography, 3D X-ray computed tomography with comprehensive image analysis and defect visualization capabilities for cracks, creep damage, grain boundary decohesion, dimensional analysis etc., X-ray diffraction, 3D profilometer, thermo-electric power and Barkhausen noise measurements.

FIG. 3. AMALIA laboratory.

JRC-Karlsruhe mainly focuses on properties of irradiated and non-irradiated nuclear fuel and materials, performing research on fuel, fuel cycle, radioactive waste, security and safeguards. A new laboratory building, known as wing M, is currently being constructed on site. Activities currently distributed among several hot laboratories of JRC Karlsruhe will be transferred into the new state of the art facility, which will contain laboratories involving the handling of highly radioactive samples of fuels and materials.

— Fuels and materials synthesis and characterisation facility (FMSC): The facility comprises 3 shielded glovebox chains for U/Th-, Pu- and Am- bearing samples, respectively. Conventional and advanced methods are available for the synthesis and characterisation of actinide compounds, including nuclear fuel samples;
— Hot cells (HC): 24 hot cells with different capabilities for the investigation of irradiated fuels, cladding and nuclear materials. The scientific studies cover safety-relevant properties and behaviour of nuclear fuels during irradiation and of spent fuel under normal and accident conditions. The available methods encompass non-destructive and destructive physical and chemical analyses. For the characterization of: structure and microstructure, morphology, fission products and phase distribution and properties; high temperature behaviour during severe accidents; mechanical characterization; dissolution; inventory/burnup determination; applications for closed cycle studies; leaching and corrosion behaviour for waste management/disposal studies;
— Materials research laboratories (MRL): series of unique, mostly home-built experimental installations dedicated to the study of thermodynamic and thermo-physical properties of actinide compounds and nuclear materials;
— Nuclear trace and analyses facility (NTA): set of installations for the chemical, physical and spectroscopic analysis of actinide and nuclear materials. It encompasses glove boxes equipped with mass spectrometers, titration chain, elemental analysis, chemical separations, gamma spectrometers, alpha spectrometers, calorimeter, neutron counters and Hybrid K-edge detectors;
— Fundamental properties of actinide materials under extreme conditions (PAMEC): state-of-the-art installations designed for basic research on behaviour and properties of actinide materials. Modular surface science laboratory with a spectroscopy station allowing photoemission, atomic force microscopy, and electron scattering measurements for the characterisation of model nuclear materials. Devices for measurements of crystallographic, magnetic, electrical transport, and thermodynamic properties as well as facilities for Np-237 Mössbauer spectroscopy, and a Nuclear Magnetic Resonance configured for studies on solid radioactive compounds;
— EUSECTRA offers a unique combination of scientific expertise, specific technical infrastructure and availability of a wide range of nuclear materials, to enable training opportunities in the field of nuclear security and safeguards. Training areas for EUSECTRA include border detection, train-the-trainers,

mobile emergency response, reach-back, creation of national response plans, nuclear forensics, radiological crime scene management, nuclear security awareness and sustainability of a national nuclear security posture. It is based on the JRC facilities at the Ispra and Karlsruhe sites;
— The large geometry secondary ion mass spectrometry laboratory (LG-SIMS) laboratory is equipped with a highly sensitive mass spectrometer to detect trace quantities of uranium/plutonium in micron-sized particles collected for safeguards purposes.

FIG. 4. JRC hot-cells.

JRC-Ispra carries out research in safeguards, security and decommissioning:

— Laser laboratory for nuclear safeguards and security: Laser based systems to carry out containment and surveillance techniques for nuclear safeguards, including fingerprinting of nuclear containers, change detection, design information verification systems and outdoor verification systems;
— Advanced safeguards, measurement, monitoring and modelling laboratory: Laboratory to measure nuclear material, to monitor the operation of facilities through an extensive collection of data from multiple types of sensors, and to model the plant operations in order to be able to analyse the data collected by the monitoring system. This laboratory is thus used for testing and developing innovative integrated solutions for the implementation of safeguards in nuclear installations;
— Performance laboratory / Pulse neutron interrogation test assembly (PERLA/ PUNITA): Laboratory for the assessment and evaluation of performances for all non-destructive assay (NDA) techniques applied in the safeguards of nuclear materials. PUNITA incorporates a pulsed (D-T) neutron generator;
— Tank measurement laboratory / Solution monitoring laboratory (TAME / SML): Bulk handling facilities, which proposes challenges to the performances of inventory quantification and density characterisation;
— Sealing and identification laboratory (SILab): Laboratory for the development, testing and commissioning of security systems used for nuclear and commercial applications;
— Illicit Trafficking Radiation Assessment Programme (ITRAP). The facility is dedicated to performing tests on radiological performances of radiation detection equipment used in nuclear security. It is composed by two laboratories: the static test lab for handheld equipment and the dynamic test lab for portals.

FIG. 5. Nuclear facilities verification laboratory.

4. JRC RESEARCH ACTIVITIES IN SPENT FUEL AND HIGH LEVEL RADIOACTIVE WASTE MANAGEMENT. ACHIEVEMENTS AND CHALLENGES

The long timeframes until disposal of spent fuel and high level waste in deep geological formations is implemented require that countries with spent fuel enable extended interim storage installations that comply with the highest levels of safety. These interim facilities, based on wet or dry storage, will be needed during periods of time significantly longer than originally expected. This requires that adequate research efforts are implemented to better understand the behaviour of spent fuel and high level radioactive waste forms, both under the conditions of extended storage, and during disposal, with the ultimate goal of providing scientific and technical evidence in support of the best suited options in terms of safety and efficiency of future spent fuel management procedures. The JRC has more than 20 years of experience in research aspects of spent fuel and high level radioactive waste management.

Understanding the impact of long term storage on properties and behaviour of spent fuel and high level radioactive waste forms to be expected during the later stages of management prior to disposal, such as for example handling, recondition (repackaging), and transport is key in terms of safety. Understanding its behaviour after disposal will also help reducing uncertainties in the assessment of the deep disposal facilities.

The safety assessment of extended storage requires defining/extrapolating the behaviour of the fuel assemblies and the package systems over a correspondingly long timescale, to ensure that the mechanical integrity and the required level of functionality of all components of the containment system are retained. Investigations on packages stored for relatively short term revealed no alterations negatively affecting the integrity of the dry storage system including spent fuel and containers [5, 6]. Since no direct measurement of 'old' fuel and/or packages can cover the ageing time of interest, such measurements must be complemented by studies aimed at targeting specific aspects and processes expected to affect properties and behaviour of spent fuel during extended dry storage. For instance, tests conducted under accelerated conditions or other relevant simulations can be useful to define the boundary conditions for the safe implementation of extended storage concepts.

During storage, radioactive decay events determine the overall conditions of the fuel and generate heat that must be dissipated. Alpha-decay damage and He accumulation are the key process affecting the evolution of properties and behaviour of spent fuel. The dose rates and the temperatures experienced during storage are lower than during in-pile operation: however, the duration of the storage is much longer (if spent fuel disposal in the repository is considered, the time interval in which radiation damage accumulates ultimately is open-ended).

The effects of alpha-decay damage and helium build-up during spent nuclear fuel storage are the object of a multi-year programme of studies carried out at JRC-Karlsruhe, which covers in particular the evolution of physical-chemical and mechanical properties [7–9] as a function of accumulated radiation/decay damage and He. The experimental characterization covers microstructure alterations, lattice swelling, thermal diffusivity, calorimetry, hardness and mechanical fracture behaviour. Irradiated LWR fuels (UO_2 and MOX) and tailor-made materials are studied. The superimposition of alpha-decay effects occurring during storage at relatively low temperature on the fuel configuration as determined by in-pile irradiation is evaluated. The investigations address processes and mechanisms from the microstructural level (lattice defects, He bubbles) up to the macroscopic properties (swelling, impact load resistance), which determine the safety performance of the spent fuel rod during long term storage. The final goal of these studies is to contribute to assessing the mechanical integrity of spent nuclear fuel rods during and after extended dry storage.

The approach combines different experimental techniques, encompassing a multiscale range from the microstructure up to the macroscopic property level. The studies are performed using irradiated fuel and tailor-made materials which allow studying alpha-decay and helium accumulation effects under accelerated ageing conditions. The trends over time/cumulative decay damage of several properties could be validated by comparing spent fuel and accelerated ageing analogues. For instance, comparative studies between spent fuel and analogues show an almost complete similarity of the basic recovery mechanisms associated with thermal annealing of alpha-decay induced defects and with He release from the fuel. Similar validation of the accelerated ageing approach could be obtained for thermal conductivity and hardening, which show satisfactory similarity between accelerated ageing analogue and spent fuel. These results indicate that these properties should not be cause for concern in case of extended spent fuel storage. The validation of the swelling trend is still under study. The radiation damage and helium generation range relevant for UO_2 up to medium-high burnup stored for 100 years may induce a lattice swelling within tolerable levels. However, analogue samples results indicate that saturation may occur at higher swelling level (up to ~0.4% for accumulated damage levels > 1 dpa). If verified in spent fuel, such swelling levels may be relevant for very high burnup UO_2 or for MOX fuel during extended storage of the order of a century. The application of these findings to spent fuel requires factoring in specific characteristics of irradiated fuel, namely its heterogeneity, which may play an overall benign role in maintaining a satisfactory degree of mechanical integrity for spent fuel.

The basic property studies are complemented by integral macroscopic spent fuel rod characterization aimed at determining safety relevant aspect which would affect the behaviour during accident conditions. Both fuel and cladding are subject of these investigations.

The fracture and fuel dispersion of LWR spent fuel rod segments subjected to simulated impact loading has been characterized experimentally at JRC-Karlsruhe in the frame of a collaboration with GNS (Germany) and AREVA (now Framatome) [10, 11] and in subsequent campaigns [12]. In this first set of tests a falling hammer device was used to test UO_2 fuel rodlets with a burnup ranging between 19 and ~74 GWd/tHM. Figure 7 shows photogram from the test performed on the ~67 GWd/tHM PWR rodlet recorded by a high speed camera placed outside the hot cell [12].

Remarkable similarities were observed among all rodlets tested, in spite of the burnup range affecting the samples tested; in particular, the amount of fuel released per fracture is similar among the samples. In all the tests the released fuel collected at the bottom of the device corresponded to ≤ 2g per fracture. Neither extensive fuel release nor special fuel release effects associated with the presence of the high burnup structure [13] were observed for the high burnup samples.

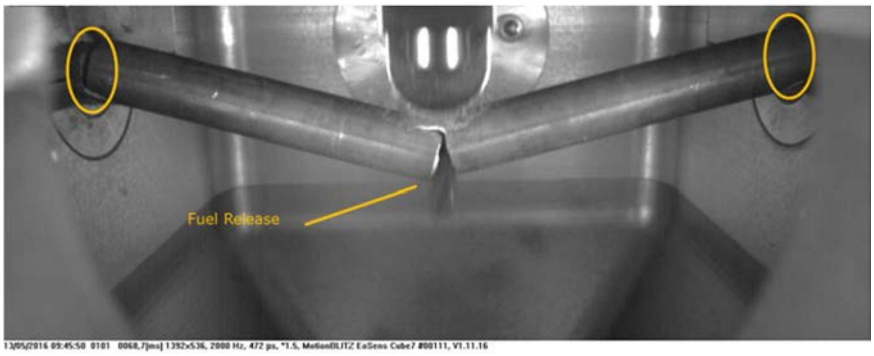

FIG. 6. High speed camera photogram illustrating the impact fracture of a ~67 GWD/tHM PWR fuel rodlet [12].

The testing campaigns are continuing and include also bending tests and other methods to determine the resistance of aged spent fuel rods against mechanical loads and the overall mechanical integrity of the spent fuel during and after extended storage. Key factors that may influence the mechanical stability, and which are specifically investigated include (high) burnup, irradiation and post-irradiation history, type of fuel (e.g. MOX), and hydride distribution/orientation in the zircaloy cladding.

The long term corrosion behaviour of spent fuel exposed to groundwater in a geologic repository is also object of multi-year study campaigns in JRC Karlsruhe hot cells. Although the combination of natural and engineered barriers will provide full sequestration of the spent fuel/high level waste from groundwater and other environmental agents, it is expected that in the remote future there will be contact between spent fuel and groundwater. Specific research topics include research and assessment of spent fuel stability and radionuclides mobilization when in contact with aqueous media. In particular, the aim of the current research projects is to investigate specific aspects of the so-called Instant Release Fraction (the fraction of radionuclides inventory available for relatively fast release upon 'first contact' between spent fuel and groundwater) for different compositions of spent fuel such as UO_2, MOX, and fuel with additives, as well as for different irradiation histories (different burnups). JRC research also cover basic processes and mechanisms, such as the factors and mechanisms determining the corrosion of the UO_2 matrix in groundwater, e.g. effects associated with expected fuel properties at the time of groundwater interaction, and effects associated with the local environment, e.g. the presence of hydrogen overpressure. The very long term fuel structure stability as a function of self-irradiation damage, in dry and wet conditions and the investigation of individual processes (e.g. affecting dissolution and re-precipitation) at the surface of the spent fuel or high level waste form are also investigated [14].

5. CONCLUSIONS AND WAY FORWARD

A few Member States in the European Union consider the spent fuel as a valuable resource and opt for a management policy of reprocessing and reusing the fissile and fertile material recovered and disposing of the high level radioactive waste resulting from the process. The spent fuel management policy of the majority of the Member States is direct disposal, although a few of these keep the option of reprocessing open and plan to take the final decision in the future.

One of the important topics of the Euratom Research and Training Programmes continues to be spent fuel management. Complementary to the research in this topic carried out by EU Member States the JRC follows and adapts to the evolution of the scenario: hence, it focuses its research on extended storage, and on reducing uncertainties of the behaviour of spent fuel under disposal conditions. To this end, JRC makes use of its research infrastructure (which can be accessed by students and researchers through several programmes to be further expanded), know-how and competences.

Regarding storage, the results so far indicate that the main mechanism that may affect properties of spent fuel is alpha-decay and He accumulation. The ongoing research on the expected evolution of some of these properties, as well as the influence of the heterogeneities of the fuel will further address processes and mechanisms from the microstructural level up to the macroscopic properties, which determine the safety performance of the spent fuel rod during long term storage. Accident condition testing (impact, bending tests) so far indicate that there is no extensive fuel release in case of spent fuel rod failure, being rather independent of the burnup. More tests will extend the database and will combine conditioning to try and reproduce properties after extended storage.

On disposal, the results so far contribute to the determination of the 'instant release fraction' for different types of fuel, different burnup, and different irradiation history; additionally, different fuel regions have been and are tested to take account of the different conditions.

JRC research work will continue in partnership with our EU and international partners with the aim of completing the work of finding the evolution in the long term of the properties of the spent fuel important for safety. It will follow and try to anticipate the evolution of the scenarios and the priorities and will further exploit the synergies among its different nuclear research lines, specifically (but not exclusively), with the ones on the fuel cycle, radioactive waste, nuclear data, and partitioning and transmutation.

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Paper ID#101
SPANISH NATIONAL STRATEGY FOR SPENT FUEL AND HIGH LEVEL WASTE MANAGEMENT.
Considerations for a Centralized Storage Facility (CSF)

F. LENTIJO
ENRESA
Madrid, Spain
Email: fler@enresa.es

P. GALLEGO
ENRESA
Madrid, Spain
Email: pgab@enresa.es

Abstract

The paper covers the Spanish national policy and strategy for spent nuclear fuel, high level and special radioactive waste management. The existing legal framework in Spain establishes the need to keep a General Radioactive Waste Plan (GRWP) up to date. The basic strategy for the management of the spent fuel (SF), high level waste (HLW) and special waste (SW) aims for their future disposal in a deep geological repository (DGR). Such stage will be preceded by a temporary storage in a centralized facility (CSF). As this is not yet available, some actions have been performed in the NPPs to avoid the saturation of spent fuel storage pools and to allow, in this way, that they could either continue to operate or to be dismantled.

The paper also describes the main processes of the CSF, its design criteria, safety case, and site selection procedure.

1. NATIONAL POLICY FOR RADIOACTIVE WASTE MANAGEMENT

In Spain, according to the Law 25/1964 on nuclear energy, the Government shall establish the national programme and policy on the management of radioactive waste, including spent nuclear fuel, and on the dismantling and decommissioning activities of nuclear installations, by means of the approval of the General Radioactive Waste Plan (GRWP). This Plan has to be approved by the Government, at the proposal of the Ministry that is in charge of energy policy (today, MITECO, Ministry for the Ecological Transition), when having succeeded the positive assessment from the Nuclear Safety Council (CSN) and after hearing the Autonomous Regions on urban planning and environmental related issues. Then, the approved GRWP has to be put to National Parliament for information purposes. In this framework, ENRESA, the national company entrusted with the management of radioactive waste and spent nuclear fuel, as well as with the dismantling and decommissioning of nuclear facilities, is responsible of submitting, every four years or when required by the MITECO, a review of the GRWP, taking into account scientific and technical progress, experience acquired, recommendations, lessons learned and best practices derived from the peer review process.

This GRWP addresses the strategies, the necessary actions and the technical solutions to be developed in the short, medium and long term, in order to ensure the spent fuel and radioactive waste adequate management. It includes the general objectives of the radioactive waste management policy, including the dismantling and decommissioning of nuclear facilities, the significant stages and schedules for compliance in view of the general objectives, an inventory of all SF and RW, as well as some estimates of future quantities, including those from decommissioning. It also includes concepts or plans and technical solutions for the final disposal facility, including the transport and the surveillance period, together with the means that must be used to preserve the knowledge of that installation in the long-term.

One of the key components of the GRWP is an evaluation of costs and the applicable financing regime. The Plan also addresses the R&D activities that are needed in order to apply solutions and some international references taken into account to define and stablish the national policy and strategy.

Finally, the GRWP shall also contain the criteria of transparency and public participation and, where applicable, agreements concluded with Member States or third countries on the management of spent fuel or radioactive waste, including the use of permanent disposal facilities.

In this context, the first GRWP was approved in 1987. Since then, the Plan has been updated several times and the sixth version (2006) is currently in force. The Government is now addressing the so-called "Integrated National Plan for Energy and Climate 2021–2030", where the strategic bases for energy policy will be set for the coming decades. Taking into account this strategy, ENRESA is now preparing the draft for the seventh GRWP.

2. NATIONAL FRAMEWORK: RESPONSABILITIES AND INSTITUTIONAL CONTROL

As already mentioned in the previous section, the Spanish Government is responsible for the design of the national policy on the management of radioactive waste, with ENRESA technical assistance. The Nuclear Safety Council (CSN) is the independent regulatory body that holds all nuclear safety and radiation protection competencies, which has the major role of keeping the Government and the National Parliament informed about this matter. ENRESA was constituted in 1984 as a public company to the service of the MITECO and is responsible for transport, treatment, conditioning, storage and disposal of spent fuel and radioactive waste, in addition to the operations related to the dismantling and decommissioning of nuclear and, when appropriate, radioactive facilities.

In accordance with the Spanish national legal framework, the main responsibility with respect to spent nuclear fuel and radioactive waste shall rest on those who have generated them or, where appropriate, on the holder of the authorization to whom such responsibility has been entrusted. They are obliged to establish and apply integrated management systems, including quality assurance, which give due priority to safety in the global management of spent fuel and radioactive waste, and may be subject to periodic verification.

Nuclear and radioactive facilities working with radioactive materials are obliged to have dedicated facilities for storage, transport and handling of radioactive waste. They must also take appropriate measures at all stages of management to protect people, workers and the environment adequately, both now and in the future, against radiological risks, so that the production of waste, in quantity and activity, is the lowest possible, according to the scientific practice existing at each moment.

The license holders of nuclear and radioactive facilities are also obliged to prepare and sign some technical-administrative specifications for the management of their spent nuclear fuel and radioactive waste, with a view to their acceptance and subsequent collection by ENRESA. These specifications will establish their period of validity, which will extend to the end of the life of the facilities, including the dismantling and decommissioning, or closure, of the nuclear facilities and, where appropriate, of the radioactive facilities. Said specifications have to be approved by the MITECO, with the prior report of the Nuclear Safety Council.

It is important to highlight that Spanish national framework for the management of SF and RW complies with all the requirements stablished in the Council Directive 2011/70/Euratom, and very few changes have been required for its transposition, because of the great robustness of the Spanish national legal and regulatory framework for decades.

The Spanish nuclear system for spent fuel and radioactive waste management is shown in Fig.1.

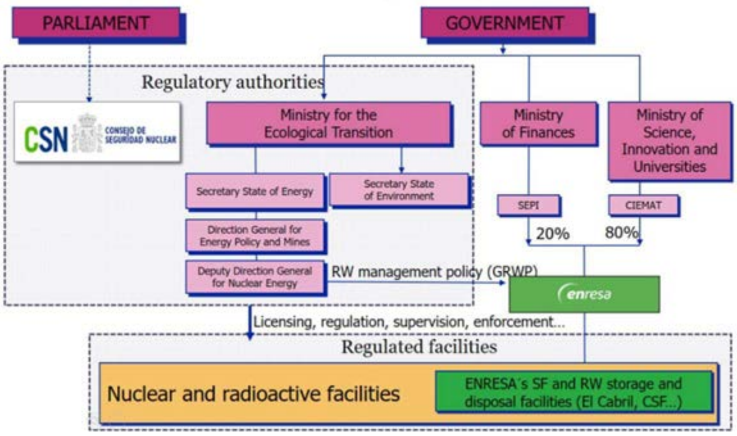

FIG. 1. Spanish nuclear system for spent fuel and radioactive waste management.

3. SPENT FUEL AND HIGH LEVEL WASTE MANAGEMENT STRATEGY, NATIONAL INVENTORY AND CURRENT SITUATION

The Spanish national strategy for the management of high level waste (HLW), included the SF, and special waste (SW), defined as the radioactive waste that does not meet the acceptance criteria of the El Cabril LILW & VLLW National Disposal Centre, aims for their future disposal in a deep geological repository (DGR). In line with the broad international consensus and with Directive 2011/70/Euratom, which recognizes that the idea generally accepted by technicians is that a deep geological disposal is the most sustainable and safe option for the SF and HLW long term management, the selected option in Spain consists of a temporary centralized storage facility, followed by a final disposal repository when available.

In this context, the Government of Spain has contemplated, in the successive GRWP since 1987, a centralized solution for the storage of SF, HLW and SW, taking into account strategic, technical, economic, safety and security considerations. This facility is named as Centralized Storage Facility (CSF). However, this project has been several times rescheduled since it was firstly conceived in 1987 as a result of the evolution and changes in the nuclear programme in Spain. The CSF licensing process started in 2014, but the Secretary of State for Energy of MITECO requested the Nuclear Safety Council (CSN), by mean of a letter issued on July 5th 2018, to temporary suspend the issuance of the mandatory report regarding the request for the construction authorization. This was decided by the incoming Spanish Government in order to analyse, in further detail, the current circumstances and carry out a more precise planning adjusted to them, which will be specified in an update of the GRWP.

In Spain there are ten nuclear power reactors: seven of them are currently in operation at five sites (Ascó I&II, Almaraz I&II, Vandellós II, Trillo and Cofrentes NPPs), one of them has been recently shutdown (Santa María de Garoña NPP), one reactor is being dismantled and decommissioned (José Cabrera NPP) and another reactor is in a safe-store period after partial initial dismantling (Vandellós I NPP).

On the one hand, as a result of the operation of these NPPs, and in accordance with the estimations of the current 6th GRWP, about 6700 tU of SF will be generated in Spain, which will result in approximately 20 000 spent fuel elements (SFE) of various types. The irradiated fuel is considered a waste since the 1983 National Energy Plan established the open cycle as the reference scenario. Additionally, due to the shipment of the spent fuel from the Vandellós I NPP to France for its reprocessing, a certain amount of HLW and SW will be returned to Spain, in the form of CSD canisters. On the other hand, as a result of the dismantling activities of the NPPs, a quantity of special waste will be generated.

Once the SF is definitively discharged from a nuclear reactor, it is stored in the on-site temporary storage in the NPPs pools, as a first step. All the Spanish NPPs are storing part or all of their SF in their pools, except for Vandellós I and José Cabrera NPPs.

As the CSF is not yet available, it has been necessary to undertake several actions in each NPP to avoid saturation of spent fuel storage pools and to allow, in this way, that they could either continue to operate or to be dismantled.

Almost all the NPPs did re-racking in their pools in different phases, consisting in the replacement of the original racks with more compact units to increase the storage capacity of the pools, based on the corresponding safety analysis. In the case of Vandellós II NPP, this re-racking was partial and the pending re-racking is expected to be completed in 2020.

Beyond the re-racking projects developed in the NPPs, the Spanish national strategy considers the construction of on-site dry storage facilities, known as individual storage facilities (ISFs), in order to ensure the continued operation of the NPPs close to pool saturation, while there is no a CSF available, or to address dismantling on termination of their operation, as stated in the 6th GRWP currently in force. Once the CSF is available, the ISFs could eventually be used as a logistic support area that would facilitate the preparation for transports. As a common rule, the ISFs are licensed as a modification of the NPP design, according to the Regulation on nuclear and radioactive facilities.

The SF, HLW and SW is stored in the ISFs by means of dry casks, that can be dual-purpose casks (conceived for both storage and transport), or storage canisters and modules that need a specific transport cask for their future transport. The casks must also be licensed, not only the design of the storage system, but also the model of transport package. These casks will be transported to the CSF, when this facility will be in operation, to be unloaded and the SF assemblies being stored at the CSF vaults (in canisters). In the meanwhile, the casks will remain at the nuclear reactors sites and its integrity must be guaranteed at any time to allow the subsequent transport and unloading of SF and HLW. During the time the NPP is in operation, these ISFs, including the casks supplied by ENRESA, are managed by the utility. Once the NPP is going to be dismantled, the site is transferred to ENRESA, which becomes responsible for its supervision, maintenance and monitoring.

In Spain, there are 6 sites where extra dry storage capacity has been developed or is being developed with the construction of ISFs:

— Trillo NPP has an ISF in operation since 2002, due to the limitations imposed by the intrinsic features of its design, where the pool has a limited capacity. It is based on a concrete building for the storage of dual-purpose metal casks with a capacity of 80 positions. 32 DPT and 2 ENUN-32P casks are currently stored in this building. The following casks expected to be stored are also ENUN-32P type;
— José Cabrera NPP was shut down in 2006 and dismantling is undergoing since 2010. The ISF is based on a concrete pad outdoors which stores steel canisters with a concrete overpack (module) that provides the necessary shielding to ensure compliance with dose limits to the public and to workers. It was commissioned in 2009 with a capacity for 16 positions. All of them are occupied for the storage of all NPP SF in 12 casks (HI-STORM) as well as the SW resulting from dismantling activities in another 4 casks (HI-SAFE), containing reactor internals and other fuel-related operational waste (attachments, pieces of structural elements for the fuel, etc.);
— These casks are only prepared for storage, although there is another cask licensed and prepared for transport (HI-STAR);
— Due to the proximity of its pools saturation capacity, the Ascó NPP (I&II) has an ISF in operation since 2013, with a similar system that the one used in José Cabrera NPP. It has a capacity of 32 casks, of which there are already 21 charged (HI-STORM type);
— Almaraz NPP has an ISF in operation since 2018, when the first ENUN-32P cask was loaded. The ISF is based on a concrete pad outdoors which stores dual-purpose metallic casks;
— The ISF at Santa María de Garoña NPP is built and licensed but not yet in operation. In 2015, the facility obtained the positive Environmental Impact Statement. The authorisation for the execution and assembly of the modification was also granted in 2015, and the authorisation for the start-up of the modification was granted in 2018. The design of this ISF, which was originally conceived for the storage of 32 casks, shall be revised after a recent Ministerial Order, denying the authorisation to extend the lifetime operation of the NPP, in order to allow the storage of the full inventory of SF. The ISF is based on a concrete pad outdoors which stores dual-purpose metallic casks. The design of these dual-purpose casks (ENUN-52B) was also approved in 2014 and the model for bulk transport in 2015, and it is expected that 5 casks to be loaded in 2019;
— The process for the licensing of the ISF of Cofrentes NPP began in 2016 with the request of the start of the Environmental Impact Assessment process. The authorisation for the execution and assembly of the NPP modification was requested in 2017. The technology is also based on a concrete pad outdoors with dual-purpose metallic casks (HI-STAR 150 type). It is expected the first cask to be loaded in 2021.

In decision-making about the characteristics of these ISFs, several variables are considered, but mainly the situation of the NPP (in operation or close to shutdown), the technologies available at that moment and the Spanish nuclear strategy development at the time of its design and licensing can be highlighted. In addition, in the selection of a specific storage system, the specificities of each NPP (maintenance means or available space) are also taken into account. For example, the ISF at Trillo NPP was conceived in the 90s when there were only preliminary plans to develop the CSF, so it was designed with a capacity that allowed the storage of the complete inventory of SF for 40 years of operation of the plant. The selected technology allowed the storage of a large number of casks (up to 80). On the other hand, in the case of José Cabrera NPP, its shutdown was scheduled at the moment of the decision, so the ISF was designed for the storage of the total inventory of the plant, including the SF from the operation and the SW that would be generated as a result of dismantling activities. The required capacity (16 casks) enforced ENRESA to select a most optimal storage technology in technical and economic terms, based on steel canisters with a concrete overpack (module) that provides the necessary shielding to ensure compliance with dose limits to the public and to workers.

The rest of ISFs were conceived and planned when the CSF Project was already initiated in design and licensing process and, for that reason, those ISFs were designed with lower capacities than the maximum required in each plant for the complete inventory of SF. In general terms, these facilities were complementary to pool capacity to reach 40 years scenario inventories. Again, in these cases, the chosen technologies were selected according to technical and economic considerations.

It is important to highlight that the design of these ISF facilities is modular to be adapted if necessary to meet the future needs of each NPP, depending on the operating scenarios that arise from the Integrated National Energy and Climate Plan, and according to the waste management strategy adopted in the planned seventh GRWP.

In any case, the solution of temporary storage of SF, HLW and SW, either in a centralized facility (CSF), or in individual facilities in each NPP (ISFs), represents an intermediate stage in the management of radioactive waste prior to the development of the Deep Geological Repository (DGR) for its definitive long term management. In this sense, ENRESA has addressed different projects in order to enhance the knowledge on the state of the art of the different technologies available for final disposal, as well as a site identification program. Additionally, ENRESA has developed its own R&D plans for decades, and will continue doing in the coming future, where several projects focused on these aspects have a major presence. Following the recommendations transferred by the ARTEMIS mission carried out by the IAEA in October 2018, ENRESA is preparing a roadmap to undertake the activities on deep geological repository to be carried out in the next decade.

4. CENTRALIZED STORAGE FACILITY (CSF)

4.1. Main functions

The centralised storage facility (CSF) shall provide a safe dry interim storage for the spent nuclear fuel (SNF) of the Spanish nuclear power plants, as well as the storage of other radioactive waste that do not meet the acceptance criteria of El Cabril disposal facility:

— Reception, process and temporary storage of the complete spent fuel coming from the operation of Spanish NPP (approximately 20 000 spent fuel elements);
— Reception, process and temporary storage of HLW coming from the reprocessing of the spent fuel of Vandellós I NPP from the French facility of La Hague (68 vitrified canisters CSD-V);
— Reception, process and temporary storage of SW (defined as them which do not meet the acceptance criteria of the El Cabril LILW & VLLW National Disposal Centre), among others.
  - Vitrified and metallic compacted waste from the reprocessing of the spent fuel of Vandellós I NPP from the French facility of La Hague (12 CSD-B and 12 CSD-C canisters);
  - Waste coming from NPPs decommissioning activities. These are activated metallic materials, mainly reactor vessel internals, or substituted BWR fuel channels;
  - Radioactive Encapsulated Sources (≈ 15 000 units).
— Reception and temporary storage of transport casks (78 positions);
— R&D activities related to the behaviour of temporary and definitive storage of SF;
— Others (cask maintenance, solid and liquid waste treatment, etc.).

4.2. Site selection

The site selection for the CSF required the development of an ad-hoc procedure based on the principles of transparency and voluntariness that included a public participation phase. In this respect, the Government created in 2006 an Inter-Ministerial Commission in charge of establishing the criteria to be met by a candidate CSF site and whose functions were to establish the reference framework with the technical, environmental and socio-economic conditions to be met by potential candidate site for the CSF, to establish and promote the process of public information, to develop the procedure by which the interested municipal areas might opt to be candidates for the site and to draw up a proposal of candidate sites to submit to the Government.

Fourteen municipalities responded to the call, although several withdrew and one did not meet all the requirements, this leaving 8 municipalities to be evaluated by the Inter-Ministerial Commission. It is worth noting that 14 420 allegations from the public and entities were formulated during the public participation process. All the process was publicly followed by a freely accessible official website, which is still available.

In September 2010, the Inter-Ministerial Commission approved a report with the sites proposal to be presented to the Government. Finally, the site to host the facility was selected in 2011.

4.3. Design criteria

The CSF design is based on a dry vault storage system for the SF and vitrified waste, while the SW is to be temporary stored in canisters at pits in a concrete building. In addition, the facility will have a temporary storage cask building (specific ISF for the CSF) in order to efficiently manage the inflow of containers from the NPPs and to temporary store casks before unloading, as well as other buildings dedicated to the storage of radioactive encapsulated sources and operational waste and a Cask Maintenance Facility (CMF) and parking area. The CSF is complemented by a Spent Fuel and HLW Laboratory (SFRWL) devoted to the study of medium and long term behaviour of these materials.

The CSF design was developed according to a generic design approved by the regulatory body in 2006, based on the operative experience of some international reference facilities upon the principle that all services are integrated in the same site.

The safety objectives of the CSF, stated in the national regulatory framework applicable to the project, establish that the facility must be designed, constructed and operated in such a way as to prevent the occurrence of accidents and, if they occur, mitigate their consequences below the normative limits of application. In order to comply with this basic safety objective, the installation is designed in such a way as to comply with the following safety functions, under normal, off-normal and accident conditions:

— Confinement: double barrier (canister, casks);
— Heating removal: passive by natural convection;
— Critically: spent fuel elements geometric disposition inside canister / casks;
— Shielding: reinforced concrete walls thickness and casks;
— Retrievability: casks and canister retrieval.

In order to ensure that safety functions are met at all operating conditions, the following design criteria will be taken into account:

— Defence in depth and multiple-barrier protection;
— Prevention of the degradation of barriers;
— Hierarchy of passive design measures;
— Double contingency in the face of criticality events;
— Failure in safe position;
— Redundancy, independence and diversity of safety systems. Single Failure Criterion;
— Application of ALARA considerations.

4.4. Layout and main processes

The CSF is divided into several buildings (Fig.2).

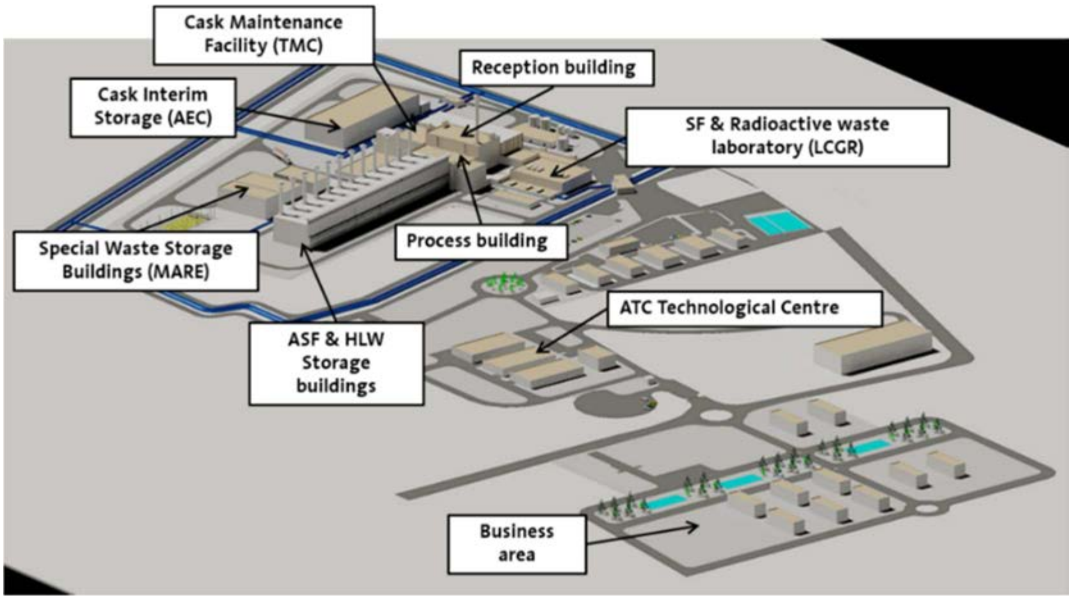

FIG. 2. General CSF layout.

The scheme of the main process is shown in (Fig. 3).

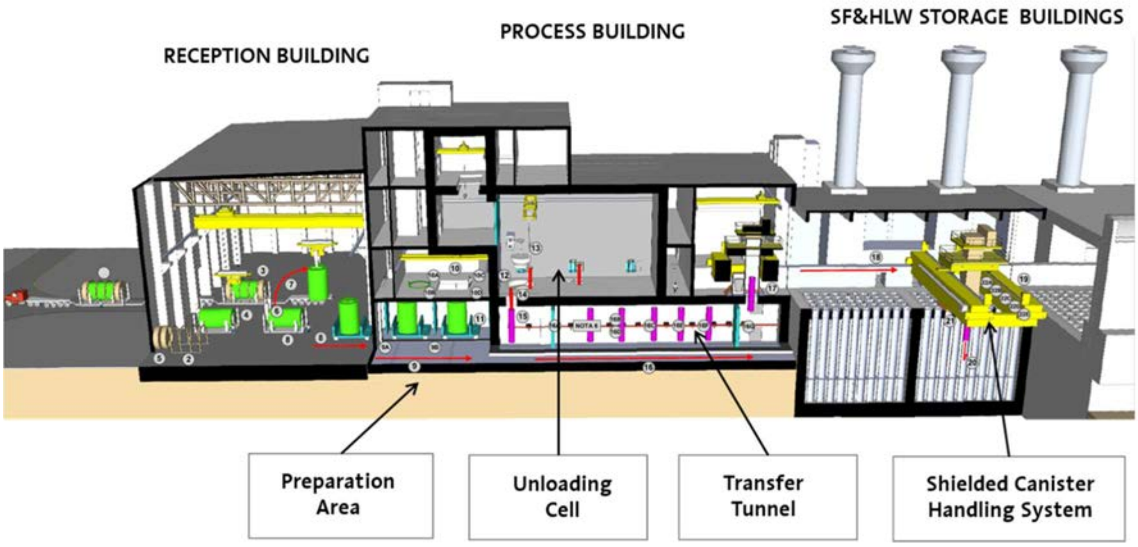

FIG. 3. CSF main process.

4.4.1. Reception building

The building has a rectangular shape and its function is to receive the vehicles that transport the casks with the radioactive waste. In this building the necessary handling and flipping operations will be carried out for the distribution of the casks with radioactive waste to the Process Building or the Spent Fuel and Radioactive Waste Laboratory, while the empty casks after its unloading will be internally transferred to the Cask Maintenance Facility (CMF), by means of a specific crane. The main operations in this building are:

(a) Loaded cask reception.
(b) Binding system removal.
(c) External contamination measurement.
(d) Visual inspection.
(e) Cask unloading.

(f) Shock absorber removal.
(g) Cask flipping into vertical position.
(h) Displacement and location in transfer trolley.

4.4.2. Process Building

This building is divided into 3 different areas:

— Preparation Area;
— Unloading Cell;
— Transfer Tunnel.

In the Preparation Area, the different mechanical processes of casks preparation before unloading the spent fuel in the hot cell will be carried out. The main operations include the control of the atmosphere in the space between the two casks lids (internal & external), unbolting and external lid removal, external contamination control (lids), internal atmosphere control, internal lid unbolting or docking adapter ring positioning. This will allow the coupling of the transport casks to the Unloading Cell in safe conditions.

In the Unloading Cell, after the cask and canister docking operations are completed, the tasks of unloading and manipulation of spent fuel loaded in the transport casks will be carried out, as well as their inspection. After removing the spent fuel elements from the transport cask, they can be positioned in the storage canisters or in the dry temporary storage pits located inside the hot cell.

When a canister is completely loaded with spent fuel elements, an internal lid is placed on it inside the Unloading Cell and it descends to the Transfer Tunnel, where the drying, inerting and sealing operations will be carried out, among others. Once all these operations are completed, the full canister will be in adequate conditions to be transferred and safely stored in the wells of the SF&HLW Storage Building.

4.4.3. Spent fuel and high level waste storage building

CSF canisters loaded with spent fuel elements in the Process Building, CSD canisters coming from the reprocessing activities (HLW or SW) and HLW canisters coming from the spent fuel and radioactive waste laboratory will be stored in vertical wells located at the bottom of the SF&HLW Storage Building, that is, in the storage vaults. This storage will be divided into 6 separate modules and will have two vaults per module (12 vaults in total). Each module will be structurally different from the next one, counting on a seismic joint between both.

The vaults will be structures with reinforced concrete walls of great thickness with independent air inlets and outlets for cooling by natural convection. In this way, the air flows through the wells and will naturally allow the evacuation of residual heat emitted by the canisters, without ever entering into contact with them.

The Management Area, common to all the vaults, will be extended over the storage vaults, through which the Canister Handling System will transit, which will transport the canisters to their storage location.

4.4.4. Special waste storage buildings

Its main function will be to store radioactive waste and packages of different types that not meet the acceptance criteria of El Cabril LILW & VLLW National Disposal Centre. To fulfil this objective, 4 different areas are designed, divided in 3 structurally different buildings:

— Operational Waste Storage (ARO). It will be a one single span reinforced concrete building which main purpose will be to store all operational wastes generated in CSF. This building will receive the Waste Management Units used for the Unloading Cell (Process Building) clogged filters and other special waste coming from maintenance and operational Unloading Cell process (lamps, MSM covers, etc.) as well as other special waste produced at CSF buildings;
— Pits Storage Building (AFO). It will be one single span reinforced concrete building which main function will be to store canisters containing metallic activated waste coming from NPPs decommissioning and dismantling activities;

- Sources Storage Building (AFU) & Extra Reserve Storage Building (ARE). It will be one single reinforced concrete frame building (two spans), which main purpose is to store all radioactive encapsulated sources (15 000 units) which will be located in shelfs or in specific areas on the floor.

4.4.5. Cask interim storage

This heavy reinforced concrete building represents a cask buffer facility to solve NPPs immediate needs of storage when there is no space at pools/ISFs and to accommodate CSF casks inflows. It is composed by independent structures, separated by seismic joints. The main areas in this building are:

- A reception area, where the casks that arrive at this facility in horizontal position in the transport vehicle will be received. In this cask loading and unloading area, the dumping platforms and the entry / exit doors of casks transport vehicles will be located;
- A transfer and maintenance area, with two transfer pits that will allow the operations in encapsulated systems (from transport systems to storage concrete modules);
- A storage area, where casks will be temporarily stored. The heat removal is ensured by natural convection (passive way) through lateral air inlets on the walls and upper air outlets on the roof;
- A decontamination zone and people control station, through which personnel can access to the outside.

4.4.6. Cask maintenance facility

It is a reinforce concrete building whose main function will be to provide the necessary facilities to carry out the maintenance and decontamination of the empty transport casks. Note that the external maintenance will be done in specific platforms located in Reception Building. Maintenance works and external and internal decontamination are planned. For this, it will consist of 3 different parts:

- Preparation Area, where the casks will be prepared for their subsequent coupling to the CMF Hot Cell. The main operations to be carried out are the external lid removal, internal lid unbolting, docking adapter ring positioning, and radiological controls;
- Hot CMF Cell, for the internal decontamination of the casks by humid way, by means of the use of steam and demineralized water;
- A Maintenance Area, for the maintenance of the interns of the casks.

4.4.7. Spent fuel and radioactive waste laboratory (LCGR)

This is a reinforced concrete building which main functions will be the study and experimentation of spent nuclear fuel and radioactive waste for the medium and long term activities. These R&D studies will be a fundamental part of the CSF project as they will support the national strategy and policies about SF&HLW management, allowing to enhance the knowledge and characterization of these materials.

The building will consist of a series of concrete cells and metal armoured cells for the study of spent fuel and radioactive waste. In addition, some glove boxes will be implemented for the works with less radioactive source term. The process will consist of the following activities, among others:

Among the operations to be performed are:

- Non-destructive tests, such as the dimensional analysis of the materials (rod length and gap), gamma spectrometry, ultrasonic inspection or induced currents tests;
- Destructive tests, such as bar cutting, pellet extraction, bar puncture and fission gas extraction;
- Specific tests, such as ceramography/metallography, mechanical tests of resilience and hardness, scanning electron microscopy, spectroscopy, mass spectrometry, etc.

4.4.8. Other buildings

The CSF project includes other facilities and auxiliary buildings, necessary to carry out the main processes and functions of the main CSF facilities: electrical building, general services building, access control building and physical security, etc.

4.5. Licensing process and safety regulatory requirements

The licensing process for both nuclear and radioactive facilities is governed by the “Regulation on nuclear and radioactive facilities” (RINR), approved by Royal Decree in 1999. These authorisations shall be granted by the Ministry for the Ecological Transition (MITECO) with the mandatory reports from CSN, which are binding when they are negative and deny authorisation and binding regarding the terms and conditions issued for the approval. These terms and conditions are part of the regulatory framework.

The request for authorization shall be submitted to MITECO, accompanied with the required supporting documentation. Afterwards, MITECO shall issue a copy of the application and the documentation to the Nuclear Safety Council (CSN) for issue of the mandatory report. This shall also be submitted to the Autonomous Communities with competencies in the area in urban and environmental planning in the territory in which the facility is to be located or the planning zone provided in basic regulation on planning of nuclear and radiological emergencies, for them to present their objections within one month. Having received the CSN report and subject to the rulings, reports and objections that may arise, MITECO shall adopt the appropriate resolution.

In this context, the CSF project licensing process is divided into 3 different steps:

4.5.1. Preliminary or siting authorization

The preliminary or siting authorisation is an official recognition of the proposal and the suitability of the chosen site. Its granting allows the licensee to begin preliminary infrastructure works that are authorised and request the authorisation for the construction of the facility. In the processing of this request, a public information period is opened, which is described in detail in answer to issue “Responsibilities for and approach to communication and public information” under this topic.

The application for preliminary authorisation must be accompanied by a site characterisation study. Other important documents are the declaration of the needs intended to be met by the facility, the draft generic design for construction and the quality assurance program.

4.5.2. Construction authorization

Entitles the licensee to initiate construction of the facility. This application is accompanied by the Preliminary Safety Analysis Report, which presents, among others, the description of the facility, the analytical radiological study and the analysis of the accidents foreseen and their consequences. Other important documents are the economic study or the previsions for dismantling and decommissioning.

During the erection of the facility and before proceeding with the loading of the fuel or nuclear material at the facility, the licensee of the authorisation is obliged to carry out a programme of pre-nuclear tests that accredits the adequate performance of the equipment or parts of the facility, both in relation to nuclear safety and radiological protection and in the applicable regulatory and technical regulations.

4.5.3. Exploitation authorization

This authorisation entitles the licensee to load nuclear fuel or to admit nuclear substances at the facility, to carry out the nuclear tests programme and to operate the facility under the conditions established in the authorisation. Initially, it will be granted provisionally until nuclear tests have been completed satisfactorily.

The application must be accompanied by the Final Safety Analysis Report, the Operating Regulation, the Operating Technical Specifications, the On-site Emergency Plan, the Radioprotection Manual and the Radioactive Waste and SF Management Plan, among others.

In 2003 ENRESA started the process by submitting to the CSN the Generic Design Safety Analysis Report, approved in 2006. After a period of site characterization and basic design, in January 2014 ENRESA submitted the application to obtain both the siting and construction authorizations. In July 2015, the CSN issued a favourable report for the siting authorization, establishing the terms and conditions to be solved by ENRESA.

On the other hand, regarding the construction authorization, work has been done on its evaluation in the period 2014–2018. However, the Ministry for the Ecological Transition, through a letter issued on July 5th, 2018 by the Secretariat of State for Energy, has recently requested the CSN to temporarily suspend the issuance of CSF project evaluation reports, in order to analyse the current situation before the official submission of the 7th GRWP.

The regulatory requirements to which the CSF facility is subject are set out in the Nuclear Safety Regulation recently issued in the national regulatory framework, which includes the principles and foundations of the European Directive 2014/87/EURATOM. In accordance with this regulatory framework, the design of the facility ensures that, during all phases of the life cycle, an adequate level of nuclear safety and radiation protection is guaranteed to the workers and to the public, avoiding the occurrence of accidents and, in case of occurrence, mitigating its consequences below the normative limits considered acceptable.

In this sense, the Preliminary Safety Analysis Report presents an analysis of compliance with safety functions in all operating conditions of the installation (normal operation, operational events and accidents), which guarantees compliance with the objectives established in Directive 2014/87.

Additionally, it is important to highlight that, during the licensing process, the regulatory body has requested ENRESA the consideration of design extension scenarios and severe conditions in the design and construction of the CSF facility. This technical instruction required the analysis of the following scenarios:

- Natural extreme phenomena hazards.
  - Seismic scenarios (seismic margin and near field earthquake).
  - Other (extreme temperatures, winds, tornados, etc.).
- Human caused extreme phenomena hazards.
  - Commercial aircraft impact;
  - Reception of a spent fuel cask with degraded safety functions.
- Total Station Blackout scenario (SBO).
- Total loss of heat sink.
- Multiple failures.

5. CONCLUSIONS AND FUTURE ACTIONS

Spain has defined its national policy on the management of spent fuel and radioactive waste for more than 3 decades, through a legal framework in which the roles and functions of each of the agents involved are clearly established. The Spanish national strategy for the management of SF, HLW and SW aims for their future disposal in a Deep Geological Repository (DGR). Such stage will be preceded by a temporary storage in a centralized facility (CSF). As this facility is not yet available, some actions have been performed in the NPPs to avoid the saturation of spent fuel storage pools and to allow, in this way, that they could either continue to operate or to dismantle, such as re-racking or the construction of some Interim Storage Facilities (ISFs).

In this context, the Government is now addressing the so-called “Integrated National Plan for Energy and Climate 2021–2030”, where the strategic bases for energy policy will be set for the coming decades. Taking into account this strategy, ENRESA is now preparing the draft for a future GRWP. In the meantime, the necessary actions are being carried out for the management of spent fuel and other radioactive waste in the NPPs that allow their operation or dismantling in safe conditions in the next years.

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FRENCH NUCLEAR FUEL CYCLE

E. TOURON
Commissariat à l'énergie atomique et aux énergies alternatives
France
Email: Emmanuel.TOURON@cea.fr

C. EVANS
Orano
France

J. VAN DER WERF
Electricité de France
France

1. BACKGROUND

As part of its efforts to help resolve the major climate and energy issues facing future generations over the next decades, France is committed to a global energy transition materialised through the Act of 17 August 2015 on the energy transition for green growth (LTECV). This act defines the main objectives for the medium and long term. Among these objectives, it is worth highlighting:

- Reduction in greenhouse gas emissions by 40% between 1990 and 2030, and a 4-fold reduction in greenhouse gas emissions between 1990 and 2050;
- Development of renewable energy sources to reach 23% of the gross final energy consumption in 2020 and then 32% in 2030;
- Reduction in nuclear energy's contribution to electricity generation to reach 50% by around 2035.

To achieve these objectives, the LTECV Act specifies the definition of a French national strategy to lower carbon emissions (SNBC) and a multi-year energy programme (PPE). The first version of this programme covers the periods 2016 to 2018 and 2018 to 2023. It must be reviewed every 5 years over a 10-year period. The main orientations of this PPE programme for the 2019–2028 period were published by the French government within the scope of a project announced in January 2019; they will be open to public consultation before their adoption scheduled for the end of summer.

a. Limiting the fraction of nuclear energy to 50% of the electricity generation.

In an effort to diversify the French energy mix to include a higher fraction of renewable energy sources and staggered investments to renew the fleet (80% of the 63 GW(e) were built in about ten years), the French government has set the objective of reducing nuclear energy to 50% of the electricity generation by 2035.

For this reason, the French government has planned to shut down 14 reactors by 2035, including the two units at Fessenheim. These shutdowns will be programmed for their fifth ten-yearly inspection outage at their latest, i.e. shutdowns between 2029 and 2035. To balance out these shutdowns over time, two reactors will be shut down pending their fifth ten-yearly inspection outage.

Faced with uncertainty about the choice of technologies making it possible to renew the nuclear fleet beyond 2035 (availability, competitiveness, environmental footprint, social acceptance, etc.), we need to maintain our skills for building new nuclear reactors based on French technology and its industrial capacity.

The French government will continue its preparation with respect to all related financial, organisational, regulatory and legal aspects before deciding whether to launch a programme to build new reactors. The conclusions of this work are expected to be ready around 2021. This programme will also focus on the management of radioactive waste produced by a new fleet of nuclear reactors.

b. Maintaining the reprocessing-recycling strategy for spent fuel

The strategy currently deployed in France is based on once-through recycling with the future objective of being able to completely close the fuel cycle by implementing the multiple recycling of spent fuel in sodium-cooled fast reactors (SFR) in the long term. France is currently one of the only countries worldwide that has mastered all of the technologies required for the treatment and recycling of spent fuel thanks to La Hague plants in La Manche department and the Melox facility in the Gard department. These plants currently employ about 10 000 staff.

Once-through recycling leads to 20 to 25% savings in natural uranium (MOX and ERU) thanks to the recycling of reusable radioactive materials (uranium and plutonium), not to mention a 4-fold reduction in the quantity of spent fuel to be stored, and a 3-fold reduction in the total volume of HLLL waste. It also leads to an improved containment of final waste. It therefore offers a number of advantages for the overall energy system and also represents an economic and industrial sector in which France boasts specific skills. For these reasons, it is important to maintain the policy of treating and recycling spent fuel in France.

2. IMPROVING THE SUSTAINABILITY OF FUEL CYCLE MANAGEMENT IN FRANCE: EMPLOYING MOX IN THE 1300 MW(E) REACTOR SERIES.

The PPE programme confirms that the strategy implemented in France will be pursued for the duration of the programme and up to 2040. For this reason and to compensate for the closure of the 900 MW(e) reactors fuelled with MOX, **a sufficient number of 1300 MW(e) reactors could be fuelled with MOX to ensure the sustainability of nuclear fuel cycle management in France. This is a short-term objective (deployment is programmed to start in the late 2020s).**

Beyond this period, the French government together with the nuclear industry will have to review its strategic orientations with respect to the fuel cycle policy and the technical options to be studied in the field of fuel cycle closure based on R&D efforts that will be continued under the PPE programme. France must continue to study all technical options that will allow it to ensure the full closure of the fuel cycle in the long term.

3. CONTINUING R&D ON FUEL CYCLE CLOSURE AND GENERATION-IV REACTORS

Up until now, research has focused on deploying the fourth generation of sodium-cooled fast reactors (SFR). Within the framework of the 2006 Act on radioactive waste management, the conceptual design of an SFR demonstrator — called ASTRID — was launched in 2010. A detailed design phase then followed between 2016 and 2019. As natural uranium resources are currently abundant and available at a low price, at least for the second half of the 21st century, it was decided that a demonstrator and the deployment of SFRs was not necessarily useful at this stage.

The SFR programme is now being reviewed and will aim at resolving the scientific and technical issues identified during the ASTRID programme studies by exploiting the knowledge and skills developed over this period. The project to build a demonstrator has thus been shelved for a later date in preparation for the commercial-scale version of these reactors expected to be built in the second half of this century. This new programme will be focusing on developing numerical simulation capabilities and on implementing a targeted experimental plan.

This new orientation in R&D sets out to consolidate and maintain our skills and knowledge in SFR physics and the related fuel cycle processes. This is a long term objective.

The level of R&D activities on the SFR fuel cycle is still high, with the development of fuel manufacturing processes using powder metallurgy by relying on past experience and by making any necessary improvements.

This research is equally important to ensure the development of a new head-end industrial pilot at La Hague to treat specific fuels such as those used in experimental reactors or those with high plutonium contents (typically but not limited to, spent fuel from fast reactors and unirradiated manufacturing scrap). This project is currently in its consolidation phase with the basic design phase expected to be launched sometime soon.

A significant part of R&D - deliberately focusing on breakthrough technologies and innovation - also aims at developing advanced processes to prepare for the construction of future fuel treatment and recycling plants for a nuclear fleet comprising a large proportion of SFRs. These processes will have to be more compact, simple, flexible, reliable and of course safer, while reducing their environmental footprint.

4. INVESTIGATING THE INDUSTRIAL FEASIBILITY OF MULTIPLE RECYCLING IN PWRS

On a shorter time-scale, multiple recycling in pressurised water reactors (PWR) could be used to stabilise our inventory of plutonium in the fleet and spent fuel, which is not the case with once-through recycling currently in place. The feasibility of this type of solution should therefore be investigated.

The question of multiple recycling in PWRs is not a new one as studies were first launched in the late 90s. Irradiated MOX fuel cannot be recycled twice with current MOX fuel technologies due to the isotopic degradation of plutonium; this calls for compensating the drop in reactivity by increasing the plutonium content in new fuel assemblies, but this requires exceeding the current authorised threshold in place for reactor safety reasons. To remedy this problem, it is possible to incorporate the additional fissile material as enriched uranium in the MOX fuel assemblies in which the plutonium content has been kept below the safety threshold.

Studies jointly carried out by the CEA, Orano, Framatome and EDF in 2017 and 2018 show that it is possible to stabilise the energy level of plutonium from a physics viewpoint, thereby authorising multiple recycling and making it possible to achieve the balance needed to stabilise the stockpiles of plutonium and spent fuels (enriched natural uranium, enriched reprocessed uranium, MOX and MOX2) using technologies that currently appear feasible. The validation of the option for a potential industrial deployment, however, necessitate the implementation of major studies and technical assessments to examine the consequences of such a solution from a technico-economic and safety perspective. Multiple recycling in PWRs is a medium-term objective with industrial commissioning deemed feasible by around 2040.

The multiple recycling of plutonium in PWRs will require the development of new fuel technologies (MIX and CORAIL). The feasibility of implementing the fuel into reactor requires an in-depth R&D programme and engineering studies. It should also be pointed out that these options generate more minor actinides and burn more plutonium which may, under certain circumstances, contribute to limit the rapid deployment of a fourth generation of nuclear reactors. Additionally, a multiple recycling strategy in PWRs will require the adaptation of fuel cycle infrastructures (adaptation of La Hague and Melox facilities or new specific workshops). From a fuel cycle perspective, some treatment and recycling operations related to these options also represent a number of scientific and technical issues that the SFR fuel cycle is also facing, fostering technological bridges between current and future fuel cycle.

A fully-costed roadmap of the project on multiple recycling, integrating both reactors and related fuel cycle aspects, is currently being drafted. It starts from R&D needs and goes up to potential industrial deployment, by going through industrial qualification steps of fuel behaviour in reactor and fuel cycle technologies. A dedicated R&D programme will make it possible to study and confirm the relevance of the various solutions for reactor safety and performances, potentially different operating conditions, factory manufacturing, transport logistics, irradiated fuel treatment. It will also include an experimental programme in reactor with the irradiation of an experimental fuel assembly sometime between 2025–2028 in view of its potential industrial deployment around 2040.

Paper ID#74
SPENT FUEL MANAGEMENT- INDIA

J.S. YADAV, K. AGARWAL
Bhabha Atomic Research Centre
Mumbai, India
Email: jsyadav@barc.gov.in

Abstract

India follows closed fuel cycle option for spent fuel management. Wet storage of spent fuel is the predominant mode of storage therefore the discharged fuel from the reactors is stored at the reactor pools which have capacity for ~10 reactor-years of operation. After appropriate cooling, the spent fuel is moved to the storage locations either on or off reactor site depending on the spent fuel management strategy. Transport of the spent fuel is carried out adhering to national and international safety guidelines in 'Type B" packages. Lower capacity fuel ponds are provided for interim storage of spent fuels at recycling facilities. PUREX process using TBP is employed for reprocessing spent fuel from PHWRs. Spent fuel reprocessing from FBRs and futuristic reactors is demonstrated using TBP based solvent extraction processes. The safe management of radioactive wastes envisages two distinct modes of final disposal in respect of radioactive wastes *viz.* near-surface engineered, extended storage for low and intermediate level radioactive wastes and deep geological disposal for high level and alpha bearing wastes. HLLW treatment is carried out in waste immobilization plants and interim storage of vitrified HLLW is carried out in solid storage and surveillance facilities. Extensive R&D in partitioning of long-lived actinides and fission products has led to the development of solvent extraction-based process flow-sheets using indigenously synthesized solvents which are deployed at engineering scale. This has resulted in the reduction of waste volume generation and extended time of repository requirements. This has also resulted in the recovery of several useful radionuclides such as ^{137}Cs, ^{90}Sr, ^{106}Ru etc. which are used for societal benefits.

1. INTRODUCTION

India has adopted three-stage nuclear energy programme [1] utilising natural uranium as fuel in thermal reactors in the first stage. Plutonium produced from spent fuel reprocessing is recycled in fast breeder reactors in the second stage where thorium is also irradiated as a blanket material to generate in-situ U-233. U-233 produced from Th fuel cycle will be used with thorium in the third stage to sustain its power generation. The three-stage nuclear power programme was originally conceptualized by Dr. Homi Jehangir Bhabha, an architect of Indian nuclear power programme, mainly because of the availability of more thorium resources than uranium in the country.

Thus, from Indian perspective, the spent fuel management by recycling is considered to be a superior option. The programme for waste management envisages disposal of low and intermediate level radioactive waste in near surface disposal facilities and reduction in radio toxicity from high level and α-bearing wastes. Waste Immobilization Plants (WIPs), employing metallic melters for vitrification of HLLW are operational in the country. SSSFs are set up for interim storage of vitrified waste. Investigations are in progress for selecting site for ultimate disposal in igneous rock formations. Extensive R&D studies on partitioning of actinides from HLLW have resulted in the development of process flow-sheets which are deployed at engineering scale employing indigenously synthesized solvents. Emphasis is given for the recovery of useful and heat generating radionuclides such as 137Cs and 90Sr. Thus, the spent fuel in India is considered as a valuable resource.

2. SPENT FUEL GENERATION IN INDIA

Currently, India has 17 operating PHWRs, 2 BWRs and 2 LWRs at various locations. Further, 6 PHWRs of 700 MW(e) are under construction stage and 10 more are in the planning stage. The operating reactors in the country are given below in Table 1.

TABLE 1. OPERATING REACTORS IN INDIA

Reactor	Type	Capacity (MW(e))
TAPS 1&2	BWR	2×160
TAPS 3&4	PHWR	2×540
RAPS 1&2	PHWR	100&200
RAPS 3-6	PHWR	4×220
MAPS 1&2	PHWR	2× 220
NAPS 1&2	PHWR	2×220
KAPS 1&2	PHWR	2×220
KGS 1-4	PHWR	4×220
KKNPP 1&2	VVER	2×1000

Under Fast Breeder Reactor (FBR) technology development programme, a 40 MW(th) Fast Breeder Test Reactor (FBTR) which was commissioned in October 1985 is operational at Kalpakkam. Experience gained from FBTR in handling the high burn up fuel has helped in designing 500 MW(e) Prototype Fast Breeder Reactor (PFBR) which is in advanced stage of commissioning. In addition to PHWRs and FBRs, two 1000 MW(e) Light Water reactors (LWRs) are in operation at Kudankulam, Tamil Nadu. Additional two such LWRs are under construction. With planned expansion of nuclear power, a challenge for spent fuel management also is being addressed. Various steps for the management of spent fuel in India are as follows:

- Spent fuel storage and its transportation;
- Wet Reprocessing - by PUREX flow-sheet;
- Spent fuel from PHWR;
- Spent fuel from FBR;
- Management of LLW, ILW and HLLW;
- Partitioning of actinides from HLLW;
- Recovery of useful radionuclide for societal applications.

3. SPENT FUEL STORAGE AND ITS TRANSPORTATION

The spent fuel storage facilities are designed, constructed and operated by following the international standards and safety guidelines. The spent fuels from the reactors are stored initially at the reactor pool. Since natural uranium is used as fuel in PHWRs, no specific configuration is required for fuel storage. The spent fuel from LWRs is stored in racks made of special stainless steel. The spent fuel is kept in the storage locations either at reactor site or away from the reactor site depending on the management strategy. In rectors under safeguard, the spent fuel is stored in reactor pools in compliance with the IAEA guidelines. Storage capacities at reactor site generally cater to store spent fuel for nearly 10 reactor-years of operation in case of PHWR fuels whereas it is about 5–7 years for LWRs. Dry storage of spent fuel is also adopted but to a much lesser extent.

Transport of the spent fuel to the storage locations are carried out adhering to IAEA safety guidelines following three lines of defence viz. the transport reliability, concept of a package and the efficacy of resources to deal with an accident. Spent fuel transport is carried out in ‘type B’ packages, designed to withstand severe accident conditions, simulated by tests, validated by approval certificates and subject to inspection. During transportation, security and safety issues are given topmost priority wherein physical security of nuclear materials from thefts, diversion etc. are covered besides exposure, contamination, criticality and environment related issues.

4. MANAGEMENT OF DAMAGED AND PREMATURE SPENT FUEL

Depending upon the degree of damage, such fuel elements are given special treatment. Fuel bundles are encapsulated safely and transported to the storage pool along with normal spent fuel for reprocessing or subjected to dry storage in special casks. The damaged fuel bundles are subjected to post irradiation examination by visual and using several non-destructive examinations viz. ultrasonic testing, eddy current testing, gamma scanning and neutron radiography for root cause analysis. Subsequently corrective measures are studied and implemented to minimize such damages to the fuels. Premature fuel is managed as normal spent fuel by reprocessing in special campaigns.

5. REPROCESSING

PUREX process has been the main workhorse of spent fuel reprocessing for the last several decades. The process utilizes 30% TBP in n-dodecane as solvent to extract uranium and plutonium from dissolved feed solution retaining bulk of fission products in the raffinate phase. Main steps involved are given below:

- Head-end treatment involving mechanical chopping of the spent fuel followed by dissolution in nitric acid.
- Feed clarification and conditioning of the feed solution for solvent extraction.
- Co-decontamination involving extraction of uranium and plutonium leaving bulk of the fission products.
- In the raffinate phase.
  - Washing/scrubbing of organic stream with nitric acid;
  - Reductive partitioning of plutonium from uranium using uranium nitrate solution;
  - Purification of uranium and plutonium streams;
  - Conversion of Pu as PuO_2 via oxalate route and uranium as U_3O_8 via ADU route;
  - Solvent wash and its recycle;
  - Waste management.

In spite of numerous advantages, TBP is found to have certain disadvantages like higher aqueous phase solubility, formation of harmful degradation products like dibutyl phosphate, third phase formation in presence of higher concentration of tetravalent metal ions viz. Pu^{4+}, Th^{4+} and Zr^{4+}, non-incinerable nature and waste management issues. Based on extensive laboratory scale studies a C, H, O and N based monoamide viz. dihexyl octanamide was proposed and investigated under simulated feed conditions [2, 3]. Extraction and stripping behaviour is found almost similar to that of TBP under identical conditions. However, it is not yet fully established for reprocessing applications using real feed solutions. Behaviour of aqueous soluble degradation product is yet to be studied. It is also noted that hydrodynamic parameters like viscosity, frothing nature etc. are also not found to be very favourable.

Even though higher homologues of TBP such as tri-n-hexyl phosphate, tri amyl and tri-iso-amyl phosphate are extensively studied [4, 5] at IGCAR for processing spent fuel from fast breeder reactors, so far only TBP is deployed for reprocessing technology in India. As of now it is a challenge for any other solvent that can replace TBP in PUREX technology.

Though India has mastered the reprocessing technology to meet the present-day requirements and future challenges, the technology is constantly being improved. Several developmental activities are being pursued to enhance the process performance. Some of the areas where R&D is being continuously pursued are:

- Improvements in the recoveries of U and Pu;
- Improved decontamination factors for U and Pu with respect to fission products;
- Improvements in partitioning techniques;
- Reduction in waste volume generation;
- Reduction in number of cycles;
- Use of non-proliferation route following co-processing and co-conversion;
- Partitioning of useful actinide and fission products for societal application.

6. WASTE MANAGEMENT

India considers HLLW as a resource rather than waste. Separation and purification of several radionuclides from HLLW is being explored for various societal applications. Solvent extraction-based engineering facilities have been deployed to partition the waste for separation of active components like U, Cs-137, Sr-90 and An-Ln using indigenously synthesized organic solvents viz. TBP, calix crown and TEHDGA in three cycles.

Cycle I: In this cycle, Sulphate Bearing High Level Liquid Waste (SB-HLLW) is first contacted with 30% TBP for recovery of uranium and plutonium. The stripped product i.e. U-rich solution from this cycle is sent back to the reprocessing plant for reuse. The lean organic is recycled.

Cycle II: After adjustment of feed acidity, the raffinate from first cycle is contacted with caesium selective calix crown 6 (CC6) solutions to extract Cs. Cs loaded in the organic phase is back extracted with water. The aqueous phase containing Cs-137 is concentrated and used to make vitrified glass pencils. Over 300 000 Ci of radio Cs has been recovered successfully so far from HLLW and used for vitrified glass pencils having specific activity of 2-5Ci/g. The vitrified pencils are supplied to BRIT, India which are being deployed for blood irradiation applications. Photograph of a typical vitrified Cs pencil is given in Fig. 1.

FIG. 1. Typical Vitrified Cs Glass Pencil.

Cycle III: The raffinate from the second cycle i.e. Cs and U lean HLLW is subjected to solvent extraction using TEHDGA, for separation/recovery of Sr and actinides. The stripped product from this cycle is predominantly rich in Sr-90 and contains actinides and lanthanides. This aqueous phase is concentrated and subsequently used as source of radio-strontium/vitrified.

The final raffinate generated from third cycle is subjected to interim storage to allow decay of short lived radio-nuclides if any and is further processed as low level waste prior to its discharge meeting the regulatory requirements. Process flow scheme deployed for the treatment of SB-HLLW at Trombay is shown in Fig. 2.

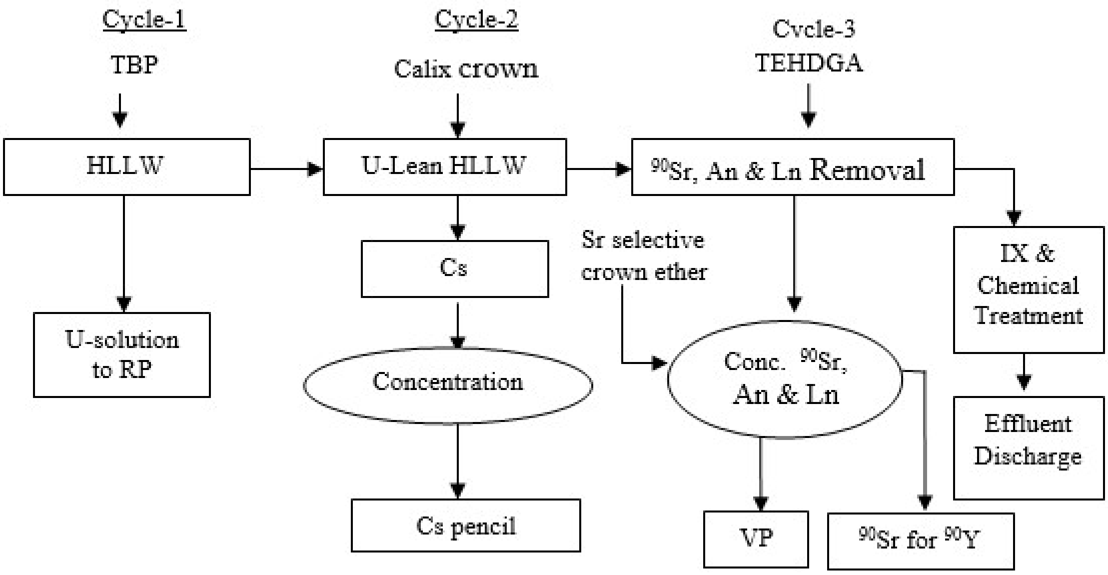

FIG. 2. Management of HLLW by Solvent Extraction Route.

The solvent extraction technology thus adopted for management of HLLW has significantly reduced the waste volumes in terms of vitrified mass and has extended the period for repository requirements.

7. SEPARATION OF Sr-90 FOR GENERATION OF CARRIER-FREE Y-90 FOR THERAPEUTIC APPLICATIONS

For achieving radiopharmaceuticals grade purity of Sr-90, multi-step separation process involving solvent extraction, ion exchange, extraction chromatography, precipitation and membrane-based techniques are deployed. Sr-90 rich stream generated from third cycle is used as feed. Sr-selective crown ether, di-(t-butyl cyclohexano)-18-Crown-6, is being synthesized and will be deployed for large scale separation of Sr-90. A two-stage SLM based generator system [6] (Fig. 3) is used for the separation of carrier free Y-90 which is principally based on the solvent extraction properties of two ligands, namely 2-ethylhexyl 2-ethylhexyl phosphonic acid (KSM-17) and octyl phenyl-N,N-diisobutylcarbamoyl methyl phosphine oxide (CMPO) under optimum conditions.

The carrier-free Y-90 acetate product lots having specific activity in the range 30-40Ci/L are supplied for Radiopharmaceutical application. To meet the higher Y-90 activity demand multiple such generator systems are under consideration.

Stage-1 *Stage-2*

FIG. 3. SLM based two stages for generation of carrier-free Y-90.

8. PARTITIONING OF ACTINIDES FROM PHWR-HLLW

An Actinide Separation Demonstration Facility was set up at Tarapur for the partitioning of actinides from PHWR-HLLW [7]. Block diagram of the integrated facilities is given in Fig. 4. In this facility, residual U and Pu are separated from the waste using PUREX solvent in first step. In the second step, actinides and lanthanides are partitioned from the waste using TEHDGA based solvent extraction process. Separation of actinides from lanthanides is proposed to be carried out using D2EHPA based process as adopted in TALSPEAK process [8] in the third step.

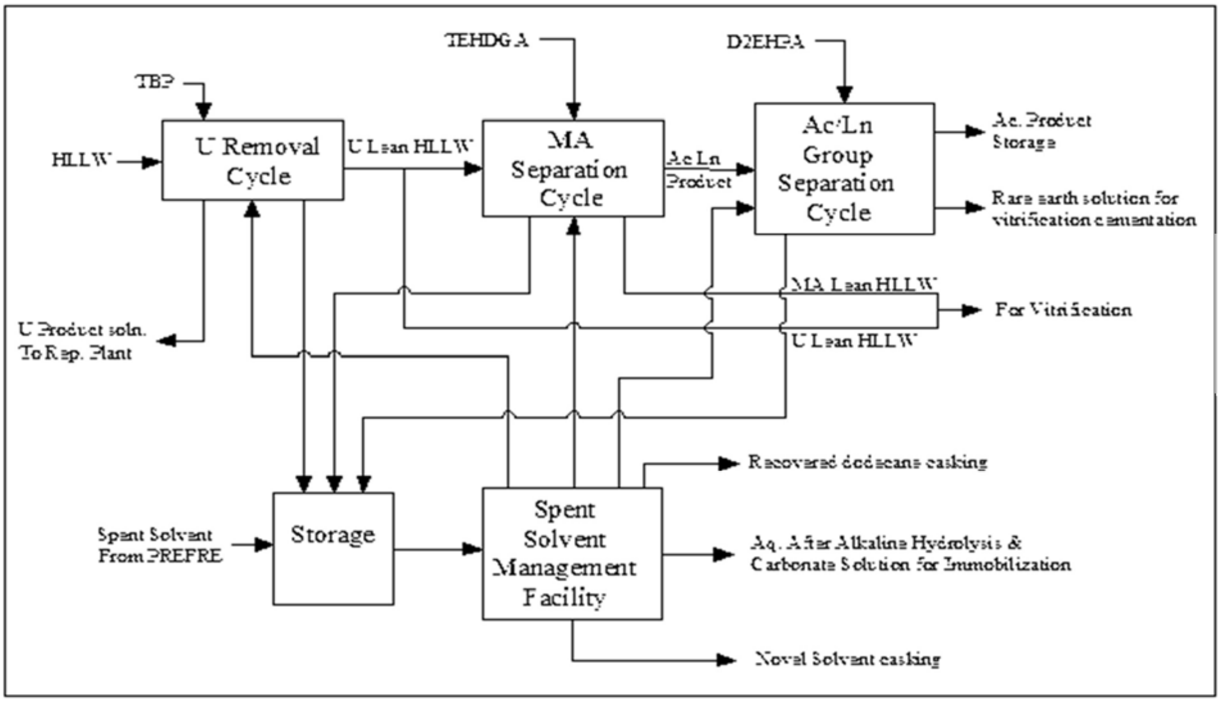

FIG. 4. Block diagram of the integrated facility for actinide partitioning.

9. CONCLUSIONS

India has been mainly operating PHWR type nuclear reactors for power generation. The country has mastered design, construction and maintenance of wet storage facilities for spent fuel meeting international safety norms. Country has also gained the experience in the design of dry storage facilities. Transportation of spent fuel from various reactors to reprocessing sites has been carried out safely following the safety requirements followed internationally. Reprocessing is carried out adopting PUREX process. High level liquid is mainly considered as resource and hence several useful radionuclides are being recovered for societal benefits. Indigenously synthesized solvents have been used for the management of HLLW. This technology has resulted in the reduction of waste volume and extended time of repository requirements. This has also resulted in the recovery of several useful radionuclides such as ^{137}Cs, ^{90}Sr, ^{90}Y, ^{106}Ru etc. which are of societal benefits. As regards to ultimate disposal, the Indian choice is focused on igneous rock formations for which evaluation of sites for repository is in progress.

ACKNOWLEDGEMENTS

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Paper ID#171
THE STRATEGY OF CLOSED NUCLEAR FUEL CYCLE BASED ON FAST REACTOR AND ITS BACK END R&D ACTIVITIES

Y. GUOAN, Z. WEIFANG, H. HUI, Z. HUA
China Institute of Atomic Energy (CIAE)
Beijing, China
Email: yeguoan@ciae.ac.cn

Abstract

To develop nuclear energy is inevitable choice for China to meet the requirement of decreasing greenhouse gas emission, at the same time of economic and society development. To ensure sustainable development of nuclear energy, closed nuclear fuel cycle strategy based on fast reactor has to be adopted. Both of recent and next R&D activities of nuclear fuel cycle back end were introduced in the paper, such as:

— Nuclear energy development and spent fuel accumulation, including fast reactor and ADS development aiming at transmutation long-lived nuclides;
— Commissioning of Reprocessing Pilot Plant for PWR spent fuel, development of advanced PUREX process and hot test of separation both U and Pu in CRARL (China reprocessing and radiochemistry laboratory);
— Minor Actinides separation on laboratory scale;
— Investigation on vitrification of high level liquid waste, high level waste disposal and its programme.

1. BACKGROUND

Nuclear units operated in China are 45 since 2018, electric power capacity is 45 GW(e), and about 8 units may be approved this year. The great demand for energy, with the development of society and economy, makes it become significant strategic option to develop nuclear power actively and efficiently for China’s energy security. According to the estimation of experts of the Chinese Academy of Engineering, nuclear capacity of China will achieve 150 GW(e) and 300 GW(e) in 2035 and 2050, respectively [1].

It is clearly clarifying that technical route of closed nuclear fuel cycle will be adopted in China. Building advanced nuclear fuel recycling system based on thermal and fast reactors combining with accelerator driven system (ADS) is the main task of next stage.

2. DEVELOPMENT OF NUCLEAR ENERGY AND NUCLEAR FUEL CYCLE

Nuclear energy development and accumulation of spent fuel is shown in Table 1, accumulated amounts of minor actinides (MA) is shown in Fig. 1. Until 2030, about 30 000 t spent fuel will be accumulated in China, including ~20 t MA; until 2050, 60 000 t spent fuel will be accumulated and amount of MA will grow up to ~55 t.

TABLE 1. NUCLEAR ENERGY DEVELOPMENT AND ACCUMULATION OF SPENT FUEL

	2020	2035	2050
Capacity/GW	58+30	150–180	300
SF/t	7000	30 000	60 000

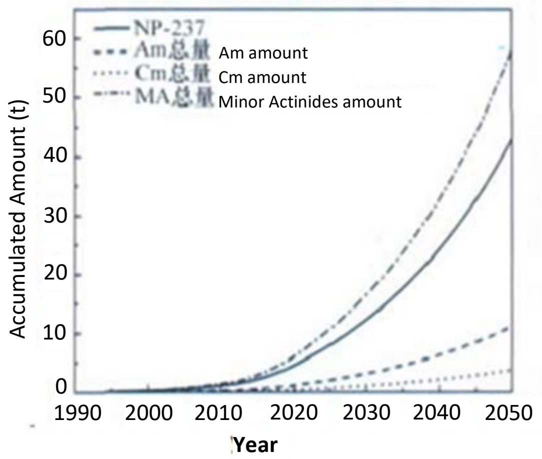

FIG. 1. Forecast for accumulated amounts of MA.

The technical development road-map of the closed fuel cycle was shown in Fig. 2 [2]. The next step is to focus on the second Generation II+ and Generation III of nuclear power to improve the safety of nuclear power systems. Besides the AP1000 in Sanmen county, disposition of independent development Generation III nuclear power Hualong One, No.3 and No.4 generating units in Fangchenggang City and No.5 and No.6 generating units in Fuqing City, are also ongoing, to further improve the Economy of Hualong one.

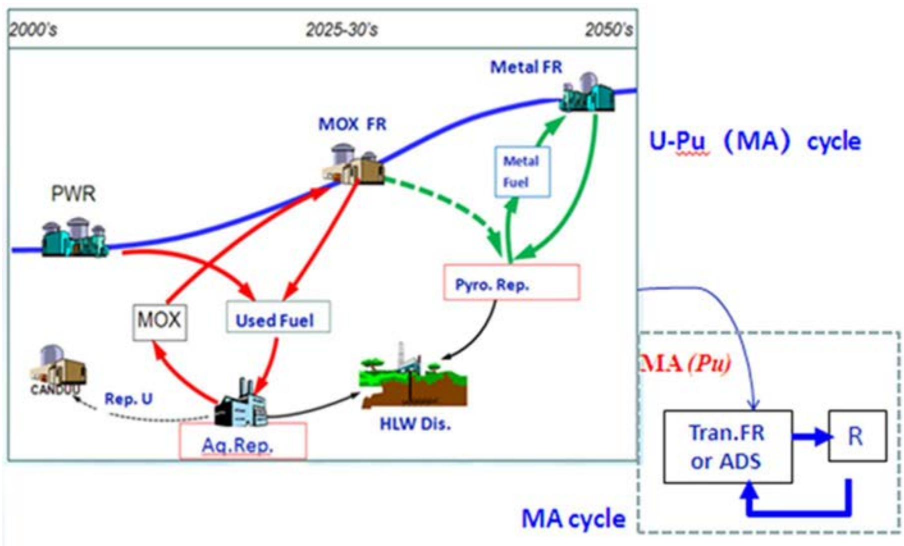

FIG. 2. Middle to long term plan to develop the closed fuel cycle in China.

To improve the utilization ratio of uranium and reduce potential long term environmental radiotoxicity hazard in radioactive wastes, FRs will be built while developing PWRs to build nuclear fuel cycle based on FR on the industrial scale during 2025–2030. From then, keeping development of FR with higher breeding ratio, meanwhile, and transmutation facility will be developed with a certain proportion. Nuclear fuel cycle based on metal fuel FR will be established around 2050, as shown in Fig. 2 [3]. Currently, the first 600 MW(e) CDFR has been approved to start construction in Xiapu, Fujian province.

At the same time, Accelerator-Driven Advanced Nuclear Energy System (ADANES) developed by CAS has been approved. ADANES consists of accelerator driven burner (ADB) and accelerator driven regeneration of used fuel (ADRUF). ADB is equal to a long refuelling cycle (16–36 years) FR with accelerator driven system, which could burn the recycled fuel containing about 50% of FPs, ADRUF could separate 50% of FPs by means of pyroprocessing separation method, the rest were converted to bred fuel, utilization ratio of fuel can reach more than 95% [4].

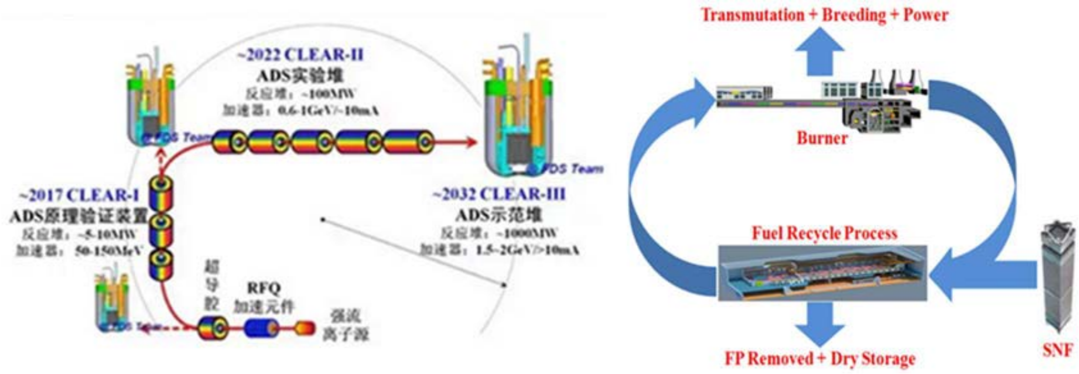

FIG. 3. Accelerator-Driven Advanced Nuclear Energy System (ADANES).

3. TRANSPORTATION AND TEMPORARY STORAGE OF SPENT FUEL

China's spent fuel implements centralized management. A 500 t spent fuel storage pools was built in reprocessing pilot plant, and a 1200 t spent fuel pools is being constructed in reprocessing demonstration plant in Jinta, Gansu province. The spent fuel transportation system from Daya Bay to Northwest of China has been established now. Spent fuel transport containers were independently developed successfully (Fig. 4).

In the following years, more spent fuel pools will be established in commercial reprocessing plant, and small scale dry storage facilities are planned to be built in Jiangsu and Guangdong province. A transportation system combining spent fuel highway-sea-rail will be further built.

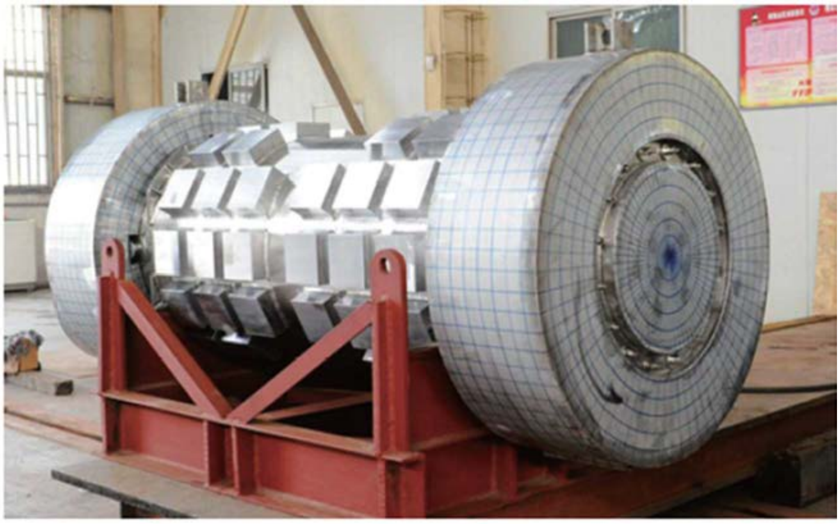

FIG. 4. 1:3 ratio of spent fuel transport container.

4. REPROCESSING ACTIVITIES AND TECHNOLOGY DEVELOPMENT

4.1. Reprocessing pilot plant

Reprocessing pilot plant (Fig. 5), which was currently in stable operation, after its hot-test carried out in 2010. Traditional PUREX process was employed in reprocessing pilot plant, 400 kg/d for head end and product finishing end, 300 kg/d for the chemical separation part, with PuO_2 and UO_3 as products. The purpose of operating reprocessing pilot plant was to demonstrate capability and stability of the used process, equipment and instrumentation under radioactive condition, and to supply plutonium for MOX fuel used in the experimental FR (CEFR).

FIG. 5. Reprocessing pilot plant.

4.2. Technology R&D

Focus on the roadmap for building nuclear fuel cycle system (Fig. 2), technical research on advanced reprocessing process, equipment, materials and analysis for spent fuel reprocessing of high burnup PWRs were mainly carried out (Fig. 6). For this reason, the government had set up National Reprocessing Technology Research Project to provide technical support for the construction of commercial reprocessing plant around 2035. At the same time, research on pyroprocessing methods are also carried out, laying the foundation for fuel cycle systems based on FR (including ADS) around 2050.

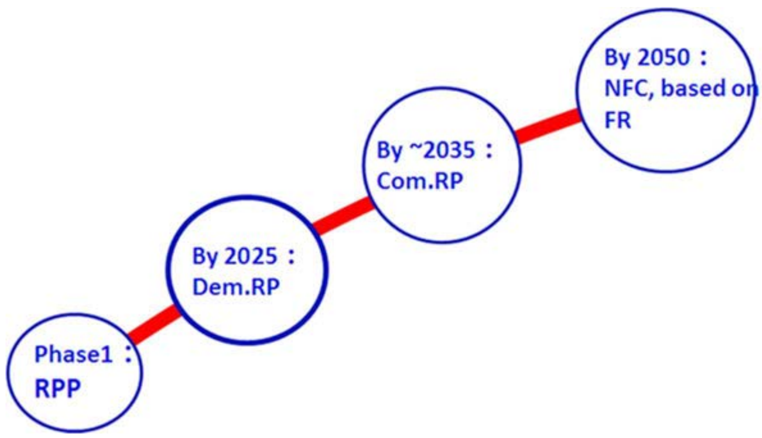

FIG. 6. Roadmap for building nuclear fuel cycle system.

4.2.1. CRARL

In order to perform hot test of reprocessing process, China Reprocessing and Radiochemistry Laboratory (CRARL) was built in Beijing in 2014 (Fig. 7) [5]. Its floor area is 10 000 m^2, including five single building as shown in Fig. 7, D block was area of hot cell complex, containing 14 hot cells. The maximum operating radioactivity was 100 000 Ci. There are more than 50 glove boxes in C block. The laboratory was put into service in 2015. Dissolution of single high burnup spent fuel rods and hot test of separation process could be operated in CRARL.

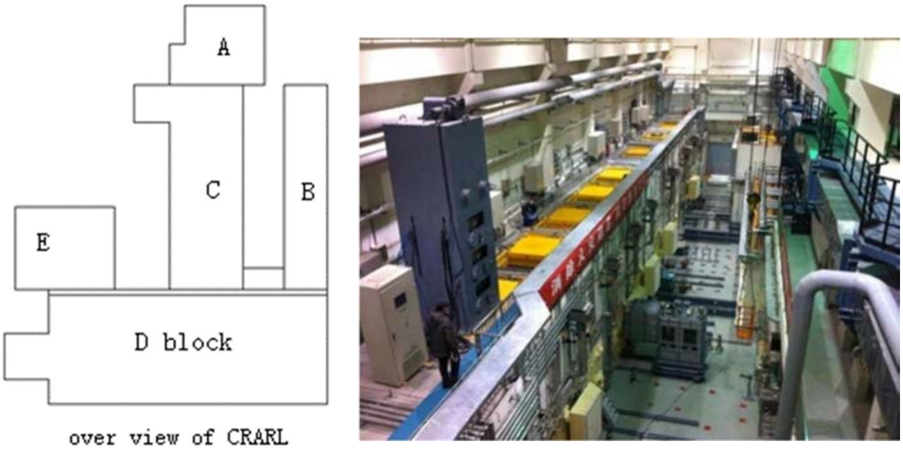

FIG. 7. Building plan of CRARL and hot cell complexes in D block.

4.2.2. Research on advanced process

In traditional PUREX process, U(IV)-N_2H_4 has been successfully demonstrated as reductants in U/Pu separation section [6]. Catalytically oxidization of U(IV)-N_2H_4 leads to the excessive use of reductants, the existence of excess N_2H_4 leads to a potential explosion risk arising from hydrazoic acid produced by N_2H_4 and HNO_2. Furthermore, U(IV) could reduce Np (VI, V) to Np(IV), leads to the dispersive distribution of Np in PUREX process.

Experiment results showed organic reductants reduce Pu(IV) at a rapid rate, reduce Np(VI,V), however at a relatively low rate [7]. Thermodynamic and kinetic studies between a series of organic reductants and Pu and Np were studied in China Institute of Atomic Energy (CIAE). Preliminary experimental results showed that N, N-Dimethylhydroxylamine (DMHAN)/monomethylhydrazine (MMH) or hydroxyl-semicarbazide (HSC) were promising reductant reagents for advanced PUREX process [8, 9].

APOR reprocessing flowsheet:

Based on the primary results, Advanced Reprocessing Process based on Organic Reductants (APOR) had been established as shown in Fig. 8, organic reductants were employed in both U/Pu separation and Pu purification cycles [10].

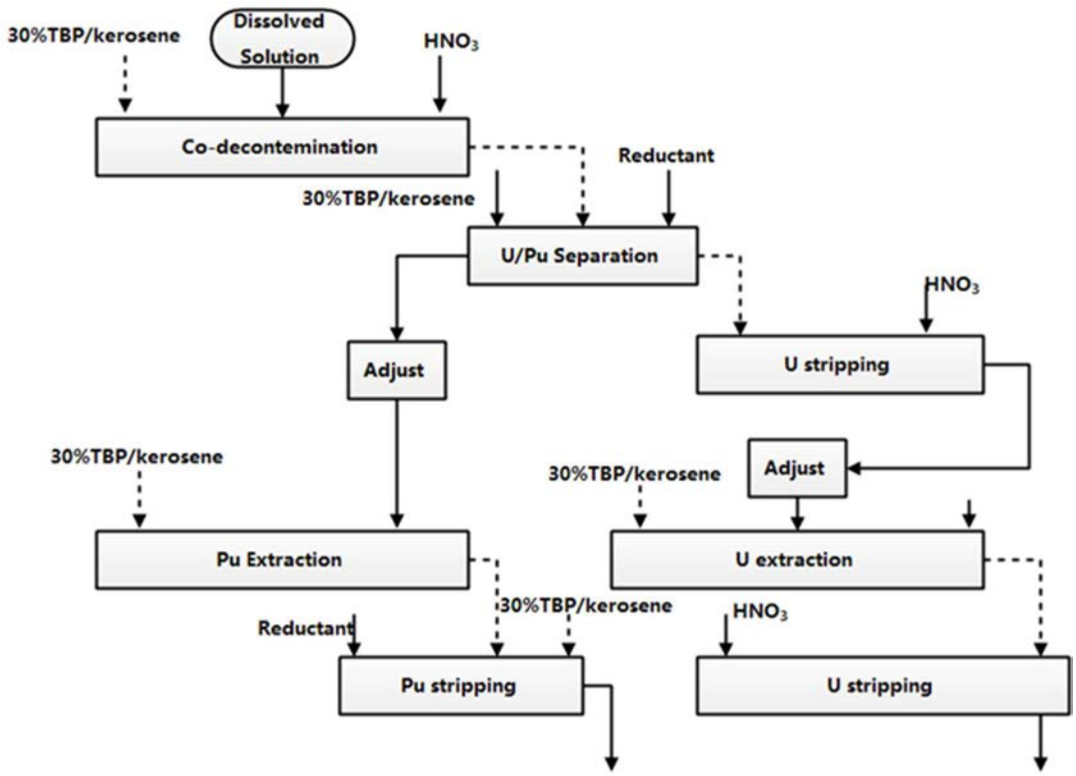

FIG. 8. The flow diagram of APOR.

Hot test of APOR process and results:

Spent fuel from research reactor (3.2% enrichment UO_2 fuel, Zr-2 alloy cladding, burnup 10 000 MWd/tU), was used. Fuel rod shearing apparatus and extraction bench are showed in Fig. 9, the APOR process showed in Fig. 8 were employed using DMHAN/MMH as reductants.

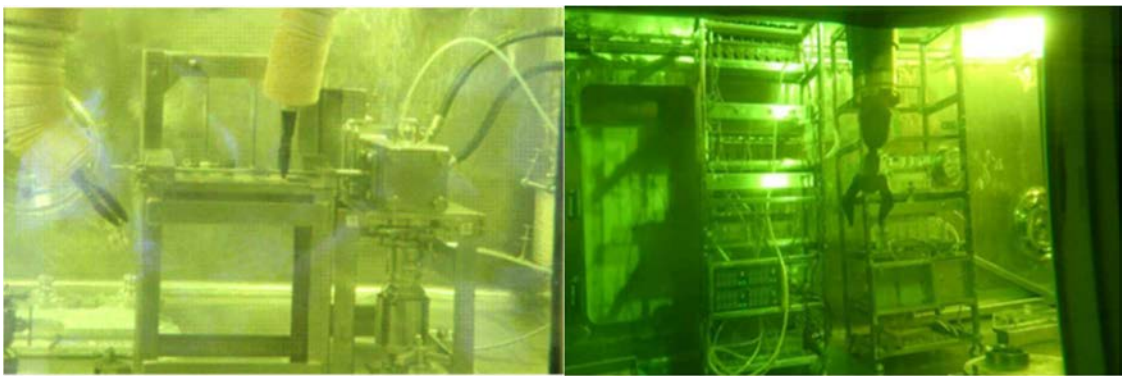

FIG. 9. Spent fuel shearing apparatus and extraction bench.

Hot tests were performed twice, total run time reached to 150 h and amount to 1.7 kgU, key technologies such as spent fuel rods transfer through sealed transfer cart, fuel dissolution and dissolution off-gas treatment, radioactive feed transfer, high efficiency separation and purification of U/Pu, preparation of over 50 analytic methods, storage of radioactive waste, control of liquid level and compressed air, environment emission monitoring and individual dose monitoring were solved during the experiment period.

Two hot tests run safely and smoothly, the experiment results achieved design targets of APOR process. Total uranium recovery rate was 99.94%, total plutonium recovery rate was 99.99%; and γ decontamination factor was more than 3.6×10^5, 1.1×10^6 for U and Pu respectively. Partial Np went to high level liquid wastes (HLLW) and most of it was controlled into Pu purification cycle and into ILLW [6].

Operating results of CRARL:

Monitoring and analysing of γ/n radioactive level, radioactive aerosol concentration, radioactive surface contamination, gas and liquid effluent, individual dose were performed, all data were lower than limits for safety analysis reports.

Monitoring of exhaust air from red area: α radioactive aerosol maintained at about 2.10×10^{-3} Bq/m^3 level, β radioactive aerosol maintained at about 2.33×10^{-2} Bq/m^3, below the management limit 6.00×10^{-2} Bq/m^3.

Monitoring of individual dose: No individual dose exceeded 0.03 mSv, all doses were lower than the management limit 2.5 mSv.

Design technology, key equipment performance indicator, safety and reliability of CRARL were comprehensively validated through the operation of hot test.

4.2.3. Minor actinides separation

TRPO process employing trialkylphosphine oxides as extractant to achieve group separation of HLLW had been established by Tsinghua University (Fig. 10) through hot test, tri-valent actinides and lanthanides in stripping solutions were separated by Cyanex301 after denitrification.

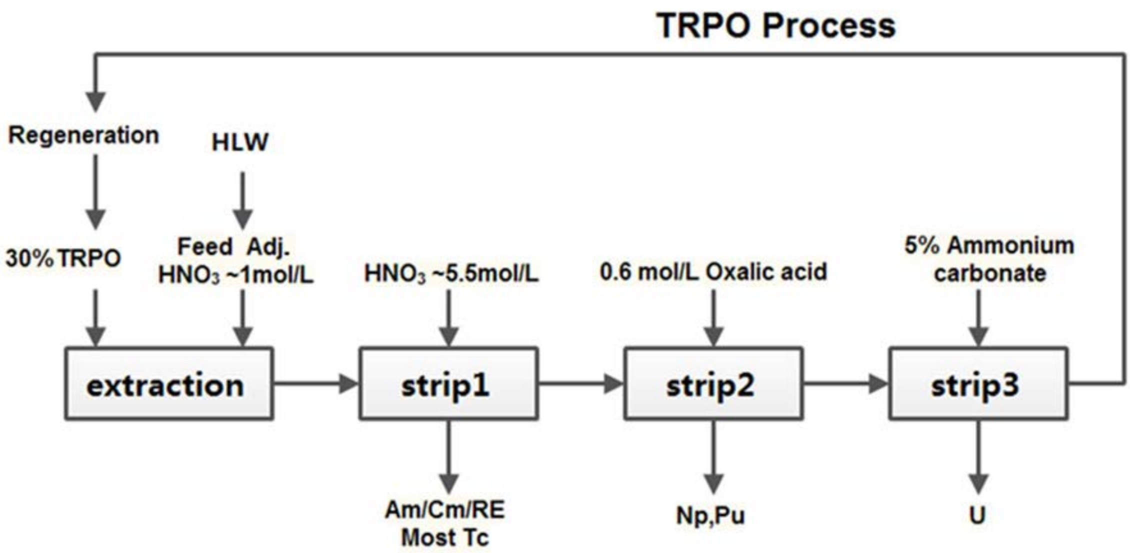

FIG. 10. The flow diagram of TRPO process [3].

HLLW group separated process using TODGA/DHOA extractants had been developed by CIAE (Fig. 11). Warm test results were showed below in Table 2 and 3, showing actinides recovery high and separation factor reasonable with ~4% remain of Am/RE in U/Np/Pu stream:

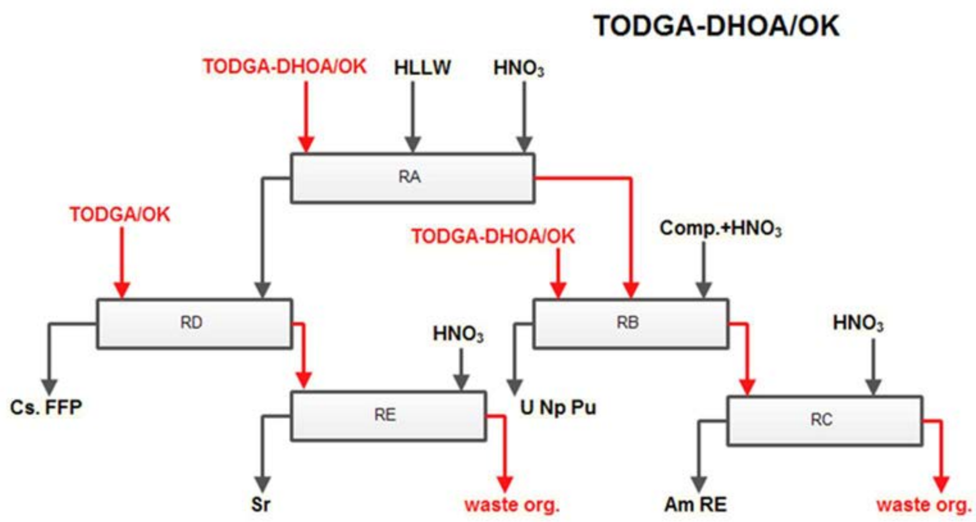

FIG. 11. The flow diagram of TODGA process.

TABLE 2. RECOVERY RATE AND MASS BALANCE OF EACH NUCLIDES IN RA EXTRACTION STAGE

Element	U	Np	Pu	Am	Eu
Recovery/%	99.98	99.99	＞99.99	＞99.99	99.91
Mass balance/%	99.60	96.28	101.23	97.60	99.24

TABLE 3. RECOVERY RATE OF EACH NUCLIDES IN RB AND RC STRIPPING STAGE

Stripping efficiencies in RB contactor %				
U	Pu	Np	Am	Eu
95.87	96.48	99.37	4.01	3.59
Stripping efficiencies in RC contactor %				
U	Pu	Np	Am	Eu
-	-	-	99.68	99.72

4.2.4. Pyroprocessing

Actinide separation over lanthanides via aluminium or gallium cathode-based electrolysis in LiCl-KCl eutectic had been carried out. Highly efficient separation of actinides over lanthanides could be achieved through forming An-Al or An-Ga alloys. Pyroprocessing separation method of classified fuels was proposed as shown in Fig. 12.

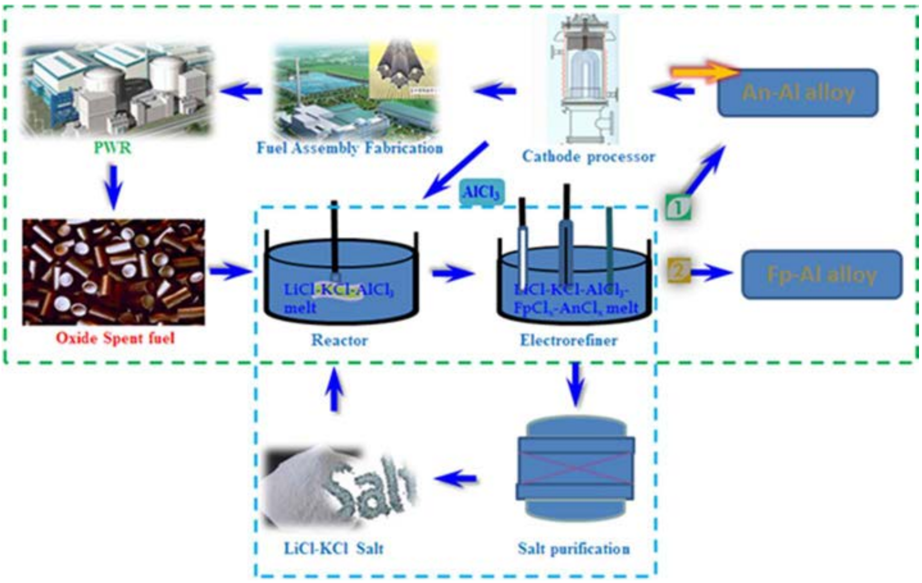

FIG. 12. The flow diagram of Pyroprocessing separation method.

5. RESEARCH OF MOX FUEL FABRICATION

An experimental MOX fuel fabrication facility [11], nearby the reprocessing pilot plant using mechanical blend (MIMAS process, experiment scale, capacity is 500 kgHM/y) has been constructed (Fig. 13). First batch of MOX fuel is under irradiation test in CDFR. Construction of a small MOX fuel fabrication plant at Northwest of China has been approved.

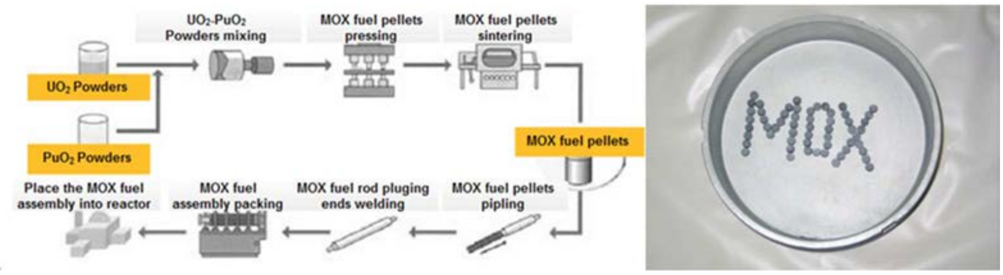

FIG. 13. Fabrication procedure of MOX Fuel assembly and 3.5%Pu-MOX pellet.

6. VITRIFICATION

First vitrification plant adopted electric furnace technology of Germany, cold test will be carried out this year in Guangyuan, old reprocessing plant location. During 2010–2014, the first principle CCM vitrification prototype was set up in China Institute of Atomic Energy. The CCM was Φ300, with its capacity 6 kg/h, max temperature around 1200°C. The bench testing was carried on continuous 24 hours, and accumulated running time was longer than 100 h. During the bench testing, several key technologies were obtained, such as start phase parameters, matching high frequency power with the melter structure, and design of drain device. From 2014～2017, the Φ500 CCM research prototype and HLLW transform system with 30 l/h were developed and set up in China institute of Atomic Energy.

7. GEOLOGICAL DISPOSAL OF HIGH LEVEL WASTE

Geological repository will be completed by 2050 [12]. Underground laboratory was planned to be built by 2026, site selection of laboratory completed. Research on security analysis of related research, engineering research, nuclide migration and disposal chemistry are being carried out (Fig. 14).

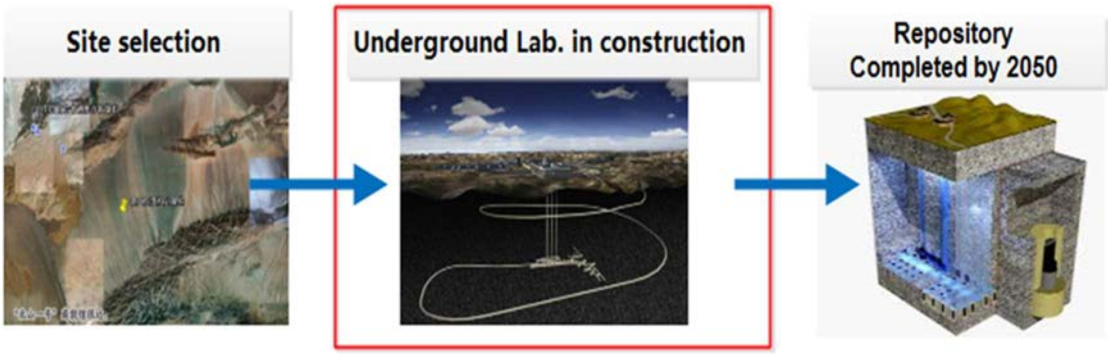

FIG. 14. 3-step strategy of repository development in China.

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LESSONS LEARNED FROM THE U.S. NATIONAL STRATEGY – A PERSONAL PERSPECTIVE

P. B. LYONS
Retired: U.S. Department of Energy Assistant Secretary for Nuclear Energy, and Commissioner of the U.S. Nuclear Regulatory Commission.
Golden, Colorado, USA
Email: peterlyons4@gmail.com

Abstract

Thirty years of watching attempts at implementation of a U.S. national strategy for high level waste management embodied in the Nuclear Waste Policy Act and its Amendments (of 1982 and 1987) from many vantage points have led to strong personal views on what has gone wrong with U.S. strategies. Instead of a repository open in 1998, the U.S. is still probably at least two decades away from opening a repository. My vantage points include management of the Los Alamos National Laboratory research programs for Yucca Mountain, years on the staff of the U.S. Senate, Commissioner of the Nuclear Regulatory Commission, and Assistant Secretary responsible for implementation of these strategies. In the talk, the stark differences between the path followed so far by the U.S. and the path recommended by the U.S. President's Blue Ribbon Commission on America's Nuclear Future will be discussed.

1. INTRODUCTION

Throughout this paper, contrasts will be drawn between two different radioactive waste geologic repository projects in the U.S. — the ongoing efforts to open Yucca Mountain (YM) and the accomplishments of the Waste Isolation Pilot Plant (WIPP). Since this paper is a personal perspective, a bit of my own history with both projects is needed, followed by very brief histories of each project.

I worked at the Los Alamos National Laboratory in New Mexico from 1969 to 2003. I was initially involved in diagnostics for underground nuclear tests, which required frequent visits to the Nevada Test Site, adjacent to YM. Later, the work at Los Alamos National Laboratory supporting R&D on YM reported through me, and I visited excavations near YM for studies on the geology and water percolation in that volcanic tuff media. And, during much of my time in New Mexico, WIPP was a major topic of discussion.

In 1997, I joined the staff of U.S. Senator Pete Domenici from New Mexico and served as Science Advisor to both him and the U.S. Senate Energy and Natural Resources Committee for eight years. WIPP and YM were significant parts of my legislative responsibilities. The Senator was responsible for the budgets of the Department of Energy (Department), and he and I travelled several times to both WIPP and YM. On visits to YM, I was struck by the amount of water in the underground environment, certainly not what I would have expected from that arid desert location. And in contrast to YM, I was impressed by the extremely dry conditions underground at WIPP.

During my tenure as Commissioner of the Nuclear Regulatory Commission (NRC) from 2005-2009, the application for YM was filed. When I was Assistant Secretary for Nuclear Energy from 2010 to 2015, I was responsible for all U.S. commercial waste management activities. The "Blue Ribbon Commission on America's Nuclear Future" [1] reported to the Department in 2012 and I directed preparation of the Administration's response, the "Strategy for the Management and Disposal of Used Nuclear Fuel and High Level Radioactive Waste", completed in 2103 [2].

2. EARLY HISTORY OF U.S. RADIOACTIVE WASTE MANAGEMENT

Extensive detail on this early history and on YM is available in the book, "The Road to Yucca Mountain," by J. Samuel Walker, former historian of the U.S. NRC [3]. Information and quotations in Sections 2 and 4 of this paper are taken from that reference.

In April 1948, J. Robert Oppenheimer, Chairman of the Atomic Energy Commission's (AEC) General Advisory Committee, dismissed the nuclear waste problem as "unimportant." But by 1955, Nobel Laureate Glenn Seaborg, who was later Chairman of the AEC, stated that "Probably the most difficult problem, which may well be the limiting factor in determining the extent to which nuclear energy will be used for industrial power, is that of disposal of the tremendous quantity of radioactive material." The U.S. National Academy of Sciences developed a report in April 1957 that stated, "radioactive waste can be disposed of in a variety of ways and at a large number of sites in the United States" and the "most promising approach for permanent disposal is to place it in salt formations". Following the guidance of the National Academy of Sciences, in 1963 the AEC directed the Oak Ridge National Laboratory (Oak Ridge) to study the suitability of an abandoned salt mine near Lyons, Kansas, called Project Salt Vault.

Large quantities of radioactive waste from the plutonium handling facility at the Rocky Flats Plant in Colorado, a part of the U.S. national defence complex, were transported annually to the Idaho National Reactor Test Station in the 1960s, but a serious fire at Rocky Flats in May 1969 focused attention on these shipments and raised environmental concerns in Idaho. To satisfy U.S. Senator Frank Church from Idaho, the AEC agreed that they would seek Congressional authorization to establish a repository for permanent disposal. With "encouraging results" reported by Oak Ridge, the AEC assured Senator Church that Idaho's wastes would be transferred to a repository that would open within a decade. However, significant concerns among the Kansas public were sparked by AEC staff comments that a decision to use the Lyons site had already been made, and the AEC purchase of land around Lyons, while seeking authorization for the entire project, further raised public fears. The project should have died quickly when the president of a nearby salt mine noted in 1971 that his shaft could channel water into Project Salt Vault and that, at a nearby injection well, about 170 000 gallons of water had mysteriously disappeared underground. The Kansas State Geologist noted that "the Lyons site is a bit like … Swiss cheese." Nevertheless, although prospects for the Lyons site were very dim by 1972, it wasn't until 1974 that the AEC Chairman officially confirmed that repository operations in Kansas were terminated.

3. THE WASTE ISOLATION PILOT PLANT

WIPP history is discussed in the book "Nuclear Reactions" by Chuck McCutcheon [4] and the brief history presented below is extracted from it.

Following the demise of Project Salt Vault, the State Senator for the Carlsbad, New Mexico region worked with local leaders to propose that the AEC study their salt beds for waste disposal. From the start, they worked with the New Mexico Congressional delegation in Washington and with the Governor of New Mexico. The local newspaper in Carlsbad was involved and maintained an open-minded editorial position. The local supporters of the project in Carlsbad studied the waste disposal plans and potential hazards and were available to discuss technical issues. The Governor's support was evident in a March 1973 message that, "As a general conclusion, I think [we] can operate under the principle that the State of New Mexico is one of the most logical locations for the national repository." An Oak Ridge report in 1972 agreed that the New Mexico part of the Permian Basin salt deposits "appears to be most promising." The initial target date for opening WIPP was 1980.

Despite the early support for WIPP, the path forward was anything but simple. Significant opposition was initially led by the Southwest Research and Information Center whose founder had a broad mistrust of nuclear power. He stated in later years that "it dawned on us that if we could make waste disposal the focus of attention, that so long as we could keep waste out of the ground, it could keep nuclear power from opening." Other groups later formed and provided further opposition to the project.

The Department aggravated concerns with repeated and confusing statements about the purpose of WIPP. Although the Department's plans were for disposal of transuranic defence waste from their national security laboratories and production sites, the Project Manager stated in 1977 that "consideration would obviously be given to making it a commercial [high level] site." Such confusion led even supporters, such as Senator Domenici, to label disposition of commercial waste in WIPP as "inappropriate and premature." Nevertheless, in December 1978, the Secretary of Energy proposed that WIPP be for purely commercial wastes. That led to a standoff with the House Armed Services Committee, which wanted the focus to stay with transuranic defence wastes. In 1979, the Department returned the WIPP site to its original mission for disposition of only transuranic defence wastes.

Development of an appropriate oversight role for New Mexico figured prominently in the history of WIPP. In 1978, the Environmental Evaluation Group (EEG) was formed in New Mexico to provide technical advice to the citizenry. The EEG was funded by a cooperative agreement with the Department, but it was a part of the New Mexico Health and Environment Department. The EEG was instrumental in dealing with WIPP's technical issues. In later years, starting in 1991, the Department funded New Mexico State University to operate the Carlsbad Environmental Monitoring & Research Center or CEMRC. CEMRC has provided independent monitoring of a wide range of environmental samples associated with WIPP. Their data, available to the public, have helped to address issues and concerns within New Mexico.

In 1979, the Chairman of the House Armed Services Committee proposed legislation for WIPP with no State participation; Senator Domenici strongly objected. They then agreed on a role for New Mexico of "consultation and cooperation." However, in 1980, the Department announced that they were moving ahead with construction to be done by 1983 - with no State involvement. The Project Manager even stated, "We don't need anything else from the State, legally or officially". An unhappy New Mexico Governor filed suit in 1981. Meetings between the Secretary of Energy and the Governor of New Mexico led to another agreement, again using the phrase "consultation and cooperation" for the State role. By then, the opening date for WIPP was listed as the end of the 1980s.

In fact, many "opening dates" were set by the Department only to be later abandoned, but they caused continued concerns in New Mexico that WIPP might open before regulatory approvals were in place. Subsequent suits were also filed, with one in 1991 based on the "obsession by the Department to get the first bins emplaced". The continued delays in opening WIPP were also of great concern in other states. For example, the Idaho Governor in 1988 imposed a temporary ban on shipments of defence waste into Idaho.

So-called "land withdrawal legislation" was required to permanently reserve the land solely for WIPP functions before certifications and waste shipments could proceed. This legislation was delayed for many years, it passed in 1992. That important legislation determined the Environmental Protection Agency (EPA) as the regulator of WIPP, barred high level waste, and provided funding for highway improvements in New Mexico. The National Academy of Sciences endorsed the safety of WIPP in October 1996, and the certification application was filed with the EPA that year. The EPA certified WIPP in May 1998, and the first waste shipment arrived in March 1999.

The performance of WIPP has not been free of issues but, except for an accident in February 2014, it has generally operated successfully. [5] That accident, which fortunately had no health consequences, was caused by improper packaging of waste at the Los Alamos National Laboratory, which led to rupture of one drum and dispersal of radioactive material in part of WIPP and in the ventilation system. The accident caused a three-year delay for cleanup and installation of a new ventilation system and cost about $500 million. WIPP reopened in January 2017 [6]. By January 2019, over 12 000 shipments had been made to WIPP and were emplaced in the salt [7].

4. YUCCA MOUNTAIN

While the growth of nuclear power in the U.S. pointed to significant issues with future waste, the early efforts to deal with that waste were complicated by shifts in government policy with regard to reprocessing. In the 1970s, the AEC focused on development of reprocessing to expedite both surface storage and geologic disposal. Proliferation concerns with reprocessing, however, led President Ford in 1976 to state that "I have concluded that the reprocessing and recycling of plutonium should not proceed unless there is sound reason to conclude that the world community can effectively overcome the associated risks of proliferation." In 1977, President Carter stated that, "we will defer indefinitely the commercial reprocessing and recycling of the plutonium produced in the U.S. nuclear power operations". In 1981, President Reagan terminated Carter's deferral.

In 1978, a Department of Energy task force estimated that a target date for opening a repository would be between 1988 to 1993. These dates greatly concerned Senator Church who was still waiting for waste to leave Idaho after progress was promised to him in 1969. To address this debate, President Carter formed an Interagency Review Group (Review Group) on Nuclear Waste Management. This Review Group made several important contributions, including citing the importance of a "waste form" that would "inhibit the release of radionuclides into water" and viewing the packaging of the waste as a way of compensating for "geologic uncertainties." The Review Group also stated that federal agencies should "interface directly and extensively with all interested and affected parties." (Those words, similar in scope to the later concept of "consent-based siting," were then subsequently ignored.)

In 1982, the Nuclear Waste Policy Act became law, requiring the Department to study at least five sites and recommend three of them to the President by 1985. The President was then to designate one site and inform Congress by 1987. Capacity of the first repository was limited to 70 000 metric tons. A second repository site was to be recommended by 1990. A surcharge on nuclear power (1 mill ($0.001) per kWh) was to be paid by generators into the Nuclear Waste Fund to finance disposal. The Department was to take possession of used fuel by 1998.

While a state governor could veto a repository chosen in his/her state, action by both Houses of Congress would override the state. Three sites were selected by the Department in 1986 in Texas, Nevada and Washington. The Department also suspended its search for a second site because its need "was not pressing".

Public concerns in the three designated areas ensued. This led to the 1987 Amendments to the Nuclear Waste Policy Act (1987 Amendments), which designated YM as the sole site for the Department's geologic characterization activities. U.S. Senator Harry Reid of Nevada (who only began his Senate tenure in 1987) promptly labelled this the "Screw Nevada Bill". The 1987 Amendments provided no path forward if YM was not successfully licensed, which supported the view that the decision to use YM was independent of technical justification. The 1987 Amendments also established the Nuclear Waste Technical Review Board to provide technical advice to Congress and the Administration; but their role is different from the EEG established for WIPP that advised the government and citizens of New Mexico.

The 1987 Amendments precipitated over 30 years of adamant opposition in Nevada. Las Vegas newspapers were strongly opposed because of YM's "proximity" to their town (about 90 miles away). Many articles claimed that transportation of high level waste in the vicinity of Las Vegas and the potential for serious accidents would destroy the gambling industry. A "Report to the Nevada Governor and Legislature" in 2000 [8] concluded that Yucca Mountain is a "bad deal for Nevada" and that YM "represents a significant gamble for Nevada's future economy and socioeconomic well-being". The Congressional delegation and State government of Nevada consistently opposed YM and raised many objections. Many "risk analyses" have been published related to YM and its impact on Nevada [9,10]. And, as the characterization of YM proceeded, it was evident that its geology was much more complex than initially thought and that the underground environment was not as dry as expected [11].

In February 2002, the Secretary of Energy recommended YM to the President. When the Nevada Governor vetoed this selection, his veto was overridden by Congress. At that time, the target opening date was 2010. In 2008, the Department submitted their YM application to the NRC and now estimated an opening date in 2020. Included in the application, in recognition of the less-than-dry conditions, was the Department's plan to place titanium drip shields over each cask of used fuel. Furthermore, these drip shields were to be put into place only at the closure of YM, about 100 years after opening, which would certainly present an interesting technical challenge. (I was an NRC Commissioner at this time, and I was rather surprised to learn that a site chosen for its excellent geological conditions would require such an extreme system of engineered barriers.) Funding for YM was stopped by President Obama in 2010, largely based on continuing opposition in Nevada [12], and it has not resumed.

The NRC, of course, analysed the repository exactly as the Department specified, i.e., with the drip shields. In 2015, the NRC issued their Safety Evaluation Report that found the Department's application generally satisfactory. The NRC staff noted that the NRC should not authorize construction until all land and water rights were in place. Issuance of the license also required successful adjudication of about 300 contentions [12, 13], i.e. issues raised by a concerned individual or group. Many of these contentions were filed by the State of Nevada.

With opposition in Nevada, it is difficult to imagine that the needed State permits will ever be granted. For example, work on YM has used water transported to the site because Nevada has never issued a water permit. Construction of the planned train route will also require many State permits. Legislation has been proposed that would remove Nevada's control over such permits, but it has not advanced beyond the House of Representatives [12].

Technical concerns with YM have been presented, which were evaluated by the NRC, including nearby seismic and volcanic activity. Perhaps the major concern involves the position of the disposal area far above the water table in a strongly oxidizing environment [14]. In such an environment, both used fuel and canister materials may not be stable in the presence of water. Some YM critics note that the U.S. is the only country considering a repository in an oxidizing environment [14]. These issues may be re-visited whenever adjudication of the contention proceeds.

5. PERSONAL PERSPECTIVES ON A SUCCESSFUL SITE SELECTION PROCESS

A comparison of the success of WIPP with the ongoing quest for YM provides a wealth of contrasts that form the basis for my personal perspectives.

5.1 Involvement of affected stakeholders with public acceptance of the repository site

At WIPP, it was local citizens acting with the government of New Mexico that proposed the AEC study of defence waste disposal at Carlsbad. Thus, the initial acceptability of WIPP within New Mexico was well developed and key stakeholders were consulted, involved and supportive.

But for YM, the 1987 Amendments simply mandated its selection with no state consultation or agreement. The 1987 Amendments supported a view that YM was chosen independent of technical feasibility and strictly by politics. The State of Nevada and its Congressional delegation have consistently fought YM ever since. (The three Nevada counties closest to YM have supported the project. However, their total population is below 2% of the State, while Clark County with the city of Las Vegas represents 73% of Nevada's population.)

While many issues have plagued the YM project, none rises to the extremes of this one. Based on my experience, consent-based siting is the only viable approach for successful completion of a repository project. Of course, an important issue for such siting is exactly whose consent is needed and what form that consent should take, and that will vary with different projects and stakeholders. But at least some significant majority of those affected by a repository choice should be supportive!

A strong lesson for the U.S. can be found in the international community. Finland, France and Sweden are moving ahead very effectively with their repository projects, each based on a consent-based process [15]. And as the "Reset of America's Nuclear Waste Management: Strategy and Policy" [16] report notes, countries like Canada, Japan and the United Kingdom that are re-evaluating their own strategies for identifying a national repository are using some variation of a consent-based process.

5.2 A management organization focused on project completion that is free from political pressures

My eighteen years in Washington taught me some of the challenges of maintaining strong federal support for a complex, decades-long, project. The U.S. political system provides opportunities for many changes in federal policy and priorities over such a long time.

Admittedly, a counter argument is that WIPP has been quite well supported despite the challenge of existing as a federal program. But WIPP grew out of strong concerns in several states with accumulation of defence wastes. Those concerns translated into several legally binding agreements between states and the federal government, with large penalty clauses for failure to move defence waste by specific dates. This provided the Department with strong fiscal motivation for achieving success at WIPP. (Not all defence nuclear waste can be accepted by WIPP, only that qualifying as transuranic and meeting strict acceptance criteria.) It was important that WIPP began with support from the New Mexico Congressional delegation, which has typically included members of both political parties, as their support has been instrumental at several key points in the history of WIPP.

But for YM, the 1987 Amendments required the federal government to create the repository and take title to the fuel by 1998, but there were no legislated financial penalties if they failed. Furthermore, once the utilities paid their fee into the Nuclear Waste Fund, their financial responsibility for the used fuel was completed - the rest of the process was up to the government. The utilities were certainly concerned when the Department failed to take title and move the used fuel by 1998, but they were then successful in suing the Department for their costs of managing used fuel at each reactor site. Thus, the utilities are not financially penalized for the absence of YM. Furthermore, while significant funds (US $6.9B through Fiscal Year 2017 [17]) have been paid to the utilities in recovering their costs, those funds are NOT derived from the Department's funds or the Nuclear Waste Fund. Instead, they are derived from the "Judgment Fund" of the Department of Treasury, which is used for paying claims against the federal government. Judgment Funds are derived from taxpayers, but the costs do not appear in the Department's budget and are not subject to annual appropriations.

In addition, the situation with the Nevada Congressional delegation was quite the reverse of the New Mexico delegation. The New Mexico delegation generally supported progress on WIPP, while the Nevada delegation was intent on blocking progress on YM. And as Senator Reid rose from the most junior Senator in

1987 to become Senate Majority Leader in 2007, the fortunes of Nevada and their ability to block funding and other legislation rose along with him.

5.3 Assured funding for the duration of the project

Any project of the magnitude and duration of YM requires assured access to adequate funding when needed by the project. Funding disruptions due to limited appropriated funds have been very damaging to the project. In contrast, the strong vested interests in many states helped keep WIPP funded and the powerful New Mexico Congressional delegation further assisted the process.

While YM is to be funded by the Nuclear Waste Fund, that Fund is not a separate bank account awaiting withdrawals. The Nuclear Waste Fund is simply part of the large federal budget and use of it is subject to appropriations just like any other federal activity. The Nuclear Waste Fund contained about US $43B in 2018 and earns about US $1.5B interest annually [17]. But, with concerns on balancing the federal budget, any proposed transfer of US $43B would have a remote chance of success. Thus, it is a daunting challenge to imagine how that Fund can be accessed today for its intended purpose. At a minimum, such access would have to be spread over many years.

In my view, if the Nuclear Waste Policy Act in 1982 had required that utilities move the fuel from their sites into safe, NRC-licensed, long term disposal by a specific date, the U.S. would probably have a functioning repository today. The utilities could have been given control over the Nuclear Waste Fund or it could have been left up to the industry to generate their own funding through rates for nuclear-generated electricity. The industry probably could have developed suitable storage and disposition sites. (Several countries have achieved or are achieving success in identifying repository sites with largely privately funded models, including Sweden, Switzerland, Finland, Canada, and France) [16].

5.4 Public education on the project

The local citizenry of Carlsbad quickly became well informed on the proposals that led to WIPP. The local paper maintained an open-minded editorial stance. State government was supportive in the early days of the project. While opposition did form later, it mostly involved groups outside of South-Eastern New Mexico. The Carlsbad supporters were always well equipped with information on the benefits and any potential hazards from the project. The State was involved in development of shipping corridors and the same is true for all states through which WIPP waste moves. Effective training of emergency responders along all WIPP transportation routes was in place before any waste moved. When there was concern with transport of waste through the capital of New Mexico, Santa Fe, a bypass route was funded by the federal government.

In contrast, YM was adamantly fought by the Nevada Congressional delegation, the State government, and the Las Vegas newspapers after the 1987 Amendments. Headlines and editorials against the project were routine. The citizens of Las Vegas were bombarded with articles [9, 10] suggesting that YM would doom the gambling industry. Fears were raised about transportation of radioactive wastes through Las Vegas. [8]

Little information to counter these local fears was presented in ways that reached most of the Nevada population. There is no question that transport of nuclear materials is handled safely throughout the world. This activity has an exemplary safety record thanks to carefully designed shipping casks and protocols. But that information was lost on the general population of Las Vegas. Furthermore, the primary route for shipment to YM proposed by the Department would involve "mostly rail" transportation utilizing a "preferred" train route from Caliente, Nevada, bypassing Las Vegas [19] – but those messages were not presented effectively. (However, concerns have been expressed by the U.S. Air Force about the choice of route by the Department of Energy [20].) The Nevada Congressional delegation also argued that transportation across the entire U.S. would present serious hazards in many states.

In addition, the Nevada delegation worked to block attempts to bring more public information into their State. When the Department proposed creation of public information resources in Nevada during my tenure on Senate staff, the Nevada delegation led by the powerful Senator Majority Leader blocked the funding. On assignment from Senator Domenici, I was sometimes asked to find paths forward on Yucca Mountain with the Nevada Congressional delegation – my discussions were far from successful.

5.6 Effective organization of a waste management campaign

Transuranic waste was routinely packed in 55-gallon steel drums throughout the Department's national security laboratory and weapons production site complex. For WIPP, shipping canisters for these drums were carefully developed and extensively tested. Video footage of some of the most dramatic tests was publicly available to demonstrate the cask's integrity under incredible accident scenarios. Today, WIPP shipments have travelled the equivalent of 30 roundtrips to the moon without a serious accident or injury. Rigid "Waste Acceptance Criteria" were developed for all waste destined for WIPP.

The commercial nuclear industry presents a dramatic contrast to WIPP. The absence of any government-mandated strategic waste management plan for all nuclear plants led to a wide variety of storage systems. Much of the used fuel is now in dry casks, but the casks vary from ones suitable and certified for transport to ones that are not. Without a disposition protocol in place for Yucca Mountain, some of the current casks might be put into YM, if they could be moved there, but others would probably require repackaging of the waste before transport or emplacement. This lack of advanced planning dramatically complicates any path forward for U.S. commercial used fuel.

Another aspect of the waste management campaign deserves discussion as well – issues of knowledge management. While the path to opening WIPP was long, it stayed within the time of a typical researcher's technical career. Thus, many of the scientists who began work on WIPP were still available for contributions as WIPP opened and in subsequent years. Continued interest in WIPP issues served to maintain funding at the lead laboratory, Sandia National Laboratories, throughout the ongoing history of WIPP. That continued funding enabled effective knowledge management, as senior staff nearing the end of their careers worked with entry-level staff to transmit their knowledge.

But the situation with Yucca Mountain is very different. Inconsistent funding of Yucca Mountain coupled with project termination and changes in operating contractors has left the knowledge base seriously fractured. Again, Sandia National Laboratories was the lead laboratory, but many of their original researchers on Yucca Mountain projects have left technical work and new staff were not always available for knowledge transfer. There have certainly been efforts to capture knowledge gained in the Yucca Mountain programme and an extensive set of literature awaits new researchers. But the invaluable ability to directly interface with the original researchers is now, in some cases, lost forever. Any effective organization for high level waste management must be sustained over decades. This need is closely coupled to the points in Subsection 5.3 discussing the need for assured funding for the duration of the project.

5.7 Technical advice for state and local governments and the citizenry

At WIPP, a matter of contention was the extent of State involvement in the disposal. While suits were filed on this issue, the outcome effectively involved New Mexico in the processes. The New Mexico EEG provided the State and local citizenry with their own technical capability to evaluate issues and the CEMRC provides independent environmental monitoring with data publicly available.

The opposition in Nevada precluded that State from forming any group like the EEG or even in seeking a strong role in project leadership. In contrast, Nevada formed the Nevada Commission on Nuclear Projects in 1985, which, in their first Report in 1986, stated that, "*The Commission … urges the Governor to continue his strong opposition*" [18]. And while the Nuclear Waste Technical Review Board was created by the 1987 Amendments, it reported to the Congress and Administration – and thus was not a resource available for trusted consultation in Nevada (and, by then, public opinion in Nevada was already firmly against YM).

6. RECENT PROPOSALS FOR SUCCESSFUL SITE SELECTION PROCESSES

My perspectives are far from unique. Many observers of the lack of progress on waste management in the U.S. have noted the same issues. Two outstanding studies have been completed in recent years exploring alternatives to the current state of U.S. high level waste management policies. Both the "Blue Ribbon Commission on America's Nuclear Future" (BRC) [1] and the "Reset of America's Nuclear Waste Management: Strategy and Policy" (Reset) [16] developed outstanding proposals to place the U.S. high level waste management program, including used fuel, on a path to success. The Administration's Strategy document in January 2013 generally supported the recommendations of the BRC [2].

The BRC and Reset studies differ in some areas. For example, the BRC recommended formation of a "Federal Corporation" to run the programme while the Reset proposed a "utility-owned, not-for-profit, implementing corporation". While I prefer the option proposed by Reset, the two studies are adamant that a new organization, separate from the Department, must be created to reform our national approach. It must isolate the project from the short-term changes in political views and it must have an assured long term funding path. Consent-based siting is prominent in both studies and, in my view, is of over-riding importance if the U.S. is to proceed toward successful management of high level waste.

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Paper ID#58
JAPAN'S NUCLEAR FUEL CYCLE POLICY

K. YOSHIMURA
Agency for Natural Resources and Energy
Tokyo, Japan
Email: yoshimura-katsumoto@meti.go.jp

Abstract

The basic policy of Japan is to promote a nuclear fuel cycle that reprocesses spent fuels and effectively utilizes the plutonium etc. retrieved, from the viewpoint of effective utilization of resources and reduction of the volume and harmfulness of high level radioactive waste. This paper explains our policy and efforts regarding the nuclear fuel cycle.

1. CONCEPT OF NUCLEAR POWER GENERATION AND DIRECTION OF POLICY

Nuclear power is an important base-load power source for Japan as a low carbon and quasi-domestic energy source, contributing to the stability of the energy supply-demand structure in the long term.

Based on this policy, on the premise that safety comes before everything else and that every possible effort is made to resolve people's concerns, judgment as to whether nuclear power plants meet the new regulatory requirements will be left to the Nuclear Regulation Authority (NRA), and in case that the NRA confirms the conformity of nuclear power plants with the new regulatory requirements, which are of the most stringent level in the world, the Government of Japan (GOJ) will follow NRA's judgment and will proceed with the restart of the nuclear power plants with the aim of realizing a power source composition ratio of 20~22% for nuclear energy in the energy mix by 2030.

In addition, accumulation of spent fuels resulting from the use of nuclear energy is a global common issue. As a responsibility of the current generation, it is essential to steadily make efforts to deal with the problems of spent fuels to avoid passing the problem on to future generations.

When taking measures, it is effective to utilize international networks, obtain cooperation from each country, and make best use of the know-how of each country obtained through research, development and dialogue activities.

2. PROMOTION OF THE NUCLEAR FUEL CYCLE POLICY

The basic policy of Japan is to promote a nuclear fuel cycle that reprocesses spent fuels and effectively utilizes the plutonium etc. retrieved, from the viewpoint of effective utilization of resources and reduction of the volume and harmfulness of high level radioactive waste.

For the research and development of fast reactors, Japan will promote R&D through international cooperation along with the Strategic Roadmap that identifies the future direction of fast reactor development.

3. THE JAPAN'S UTILIZATION OF PLUTONIUM

In July 2017, the Atomic Energy Commission decided the "Basic Principles on Japan's Utilization of Plutonium". This is our policy direction based on Japan's own initiative. It has stipulated that Japan will reduce the size of its plutonium stockpile, and, based on the realization of the following measures, the stockpile is not to increase from the current level:

— Approve reprocessing plans under the Spent Nuclear Fuel Reprocessing Implementation Act so that reprocessing is to be carried out only to an extent necessary for steady pluthermal power generation, reflecting the operational situation of the Rokkasho Reprocessing Plant (RRP), the MOX Fuel Fabrication Plant, and MOX-burning reactors;
— Instruct the operators and confirm that the produced MOX fuel is to be fully consumed in a timely manner; Instruct the operators so as to secure a balance between demand and supply of plutonium, minimize the feedstock throughout the process between reprocessing and irradiation, and reduce the feedstock to a level necessary for proper operation of the RRP and other facilities;

- Instruct the operators and confirm that the produced MOX fuel is to be fully consumed in a timely manner; Instruct the operators so as to secure a balance between demand and supply of plutonium, minimize the feedstock throughout the process between reprocessing and irradiation, and reduce the feedstock to a level necessary for proper operation of the RRP and other facilities;
- Instruct the operators and confirm that the produced MOX fuel is to be fully consumed in a timely manner; Instruct the operators so as to secure a balance between demand and supply of plutonium, minimize the feedstock throughout the process between reprocessing and irradiation, and reduce the feedstock to a level necessary for proper operation of the RRP and other facilities;
- Steadily promote efforts toward expanding storage capacity for spent fuel.

4. CURRENT STATUS OF NUCLEAR FUEL CYCLE RELATED FACILITIES

Currently Japan Nuclear Fuel Limited (JNFL), the operator for the reprocessing plant, is constructing a reprocessing plant in Rokkasho, Aomori Prefecture.

The active test of the Rokkasho Reprocessing Plant has been almost completed. Currently Japan Nuclear Fuel Limited (JNFL) is dealing with the safety review toward the completion in the first half of 2021.

As for the MOX fuel fabrication plant, JNFL is dealing with the safety review for completion in the first half of 2022.

5. ABOUT THE SPENT NUCLEAR FUEL REPROCESSING IMPLEMENTATION ACT

In 2016, the Spent Nuclear Fuel Reprocessing Implementation Act was introduced, to collect contributions from electric power companies at the time when spent fuel is generated, so that reprocessing would be carried out steadily under electricity market reforms.

Based on this law, the Nuclear Reprocessing Organization of Japan (NuRO), which is authorized by the Minister of Economy, Trade and Industry, has been established as the responsible organization of reprocessing. The organization collects contributions from the electric power companies according to the amount of spent fuel generated and consigns the reprocessing operation to JNFL.

In addition, NuRO submits a plan stipulating the amount of reprocessing to the Minister of Economy, Trade and Industry to be approved by the Minister.

As one of five measures in the "Basic Principles on Japan’s Utilization of Plutonium" decided by the Atomic Energy Commission, it has been decided to “Approve reprocessing plans under the Spent Nuclear Fuel Reprocessing Implementation Act so that reprocessing is to be carried out only to an extent necessary for steady pluthermal power generation, reflecting the operational situation of the Rokkasho Reprocessing Plant (RRP), the MOX Fuel Fabrication Plant, and MOX-burning reactors; Instruct the operators and confirm that the produced MOX fuel is to be fully consumed in a timely matter.”

The "Spent Nuclear Fuel Reprocessing Implementation Act"

- ✓ In order to ensure steady and efficient reprocessing operations under electricity market reforms, the amendment bill to the "Spent Nuclear Fuel Reprocessing Fund Act" was approved by the Diet on May 11, 2016 and promulgated on May 18, 2016.

- ✓ This act
  1) Establishes an authorized organization (NuRO: Nuclear Reprocessing Organization of Japan) responsible for reprocessing spent fuel;
  2) Secure funds for reprocessing by obligating electric power utilities to make annual payments to NuRO;
  3) Gives the Minister of Economy, Trade and Industry authorization to approve the master plan for reprocessing activities by NuRO. The Minister will approve the plan only when it is line with the policy of not possessing plutonium without specific purposes.

Scheme under Spent Fuel Reprocessing Implementation Act

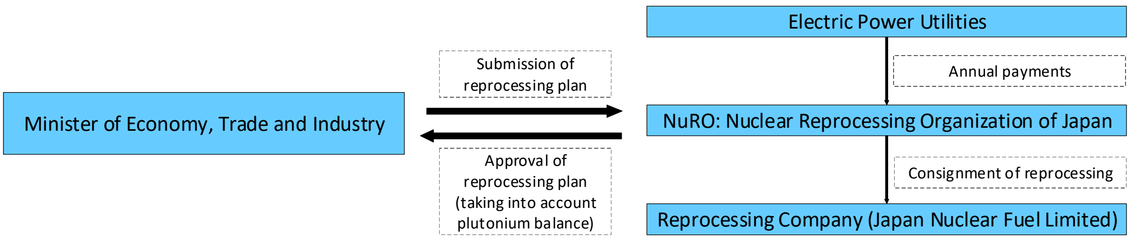

Fig. 1. Spent nuclear fuel reprocessing implementation act.

6. THE POSITION OF JAPAN TOWARDS THE DEVELOPMENT OF FAST REACTORS

Regarding the development of Fast Reactors, the government's policy is to promote R&D of fast reactors through international cooperation with the United States and France, under a Strategic Roadmap based on the Policy on Fast Reactor Development (decided by the Inter-Ministerial Council for Nuclear Power Introduction in December 2016).

In March 2017, based on the policy of fast reactor development, the Strategic Working Group to formulate a Strategic Roadmap to identify development work for the next 10 years was kicked off.

The Strategic Working Group was held 16 times in total, and in December 2018 the draft of the Strategic Roadmap was compiled.

After going through the Fast Reactor Development Conference in the same month, the Strategic Roadmap was decided at the Ministerial Conference on Atomic Energy. This Roadmap clarifies the significance of the fast reactor, such as reduction of high level radioactive waste and reduction of harmfulness, in addition to the effective utilization of resources. Having said that, the Roadmap states that the fast reactor development is a long term project, and therefore strategic flexibility is necessary in its development. It also states the policy to pursue diverse fast reactor technology while foreseeing the future possibility of nuclear technology. The Roadmap states such new policies regarding the development of a new fast reactor.

Also, the Roadmap shows that it is thought that full-scale use of the fast reactor is expected to be realized in the latter part of the 21st century, and the start of operation of the first fast reactor is expected to be at a suitable timing, for example, around the middle of the 21st century. As a way to spend 10 years in Japan for the foreseeable future, the Government of Japan decided to promote diverse technology development in 5 years and develop technologies that narrowed down in the latter half of the next five years.

Based on the Strategic Roadmap in the future, the Government of Japan will promote fast reactor development.

7. MEASURES TOWARD FINAL DISPOSAL OF HIGH LEVEL RADIOACTIVE WASTE

Finally, I will explain the current status of efforts towards the final disposal of high level radioactive waste that is generated as a result of the nuclear fuel cycle.

With regard to final disposal of high level radioactive waste, based on the Act on Final Disposal of Specified Radioactive Waste (Final Disposal Law) established in 2000, the Nuclear Waste Management Organization of Japan, which is the executing organization of high level radioactive waste was established, and three stages of survey were set up: literature survey, preliminary investigation and detailed investigation. Based on this, NUMO has started public offering of municipalities to accept the literature survey since 2002. In the midst of this process, the Government of Japan revised the Basic Policy on Final Disposal of Specified Radioactive Waste (Decided by the Cabinet in May 2015) to solve the issue of high level radioactive waste so as to not postpone burdens on future generations as the responsibility of the current generation that generated waste, and the Government decided to take the initiative.

Based on this basic policy, in July 2017, following the final disposal-related ministerial meeting, the government of Japan announced the Scientific Characteristic Map regarding final disposal, shown in the form of Japan’s country map. Taking this publication as an opportunity, GOJ will further strengthen initiatives such as promotion of diverse dialogue activities based on the interests of citizens under the collaboration of related ministries and agencies, aiming for the acceptation of the survey of disposal sites by multiple regions.

Regarding final disposal, the policy is to promote international cooperation, such as sharing dialogue methods with countries having common problems and promoting mutual utilization of domestic and international research infrastructures. In November 2018, the government of Japan held an international workshop with OECD NEA, shared the experiences of dialogue activities of each country, and shared the importance to continue learning from success stories through international collaboration.

8. THE POSITION AND CURRENT STATUS OF GOJ TOWARDS SPENT FUEL MANAGEMENT

In Japan, the basic policy is to promote the nuclear fuel cycle by reprocessing spent fuel and recycling recovered plutonium etc. For high level radioactive waste formed by vitrification of non-recyclable components remaining in the process, GOJ are promoting efforts towards their geological disposal.

Efforts towards geological disposal of high level radioactive waste require a long period of time, and it is necessary to safely manage the spent fuel generated by nuclear power generation until reprocessing is carried out.

In October 2015, the government called a meeting of the Inter-Ministerial Council for the Final Disposal of High Level Radioactive Waste and adopted the Action Plan for Spent Fuel Storage Measures. Pursuant to the plan, nuclear operators formulated plans to promote measures for the disposal of spent nuclear fuel and are proceeding with efforts to expand the capacity to store spent fuels.

Also, the Council for the Promotion of Spent Fuel Storage Measures was established in 2015 to confirm the progress of spent fuel storage measures between government and business operators. In November 2018, the 4th Council was held, and the Minister of Economy Trade and Industry asked the operators for further cooperation with the GOJ and among themselves in taking measures for spent fuel management.

9. CONCLUSION

The basic policy of Japan is to promote a nuclear fuel cycle that reprocesses spent fuels and effectively utilizes the plutonium etc. retrieved, from the viewpoint of effective utilization of resources and reduction of the volume and harmfulness of high level radioactive waste.

In July 2017, the Atomic Energy Commission decided the Basic Principles on Japan’s Utilization of Plutonium.

In 2016, the Spent Nuclear Fuel Reprocessing Implementation Act was introduced so that reprocessing would be carried out steadily under electricity market reforms. Regarding the development of Fast Reactors, the government’s policy is to promote R&D of fast reactors through international cooperation with the United States and France, under a Strategic Roadmap to be developed pursuant based on the Policy on Fast Reactor Development (decided by the Inter-Ministerial Council for Nuclear Power Introduction in December 2016).

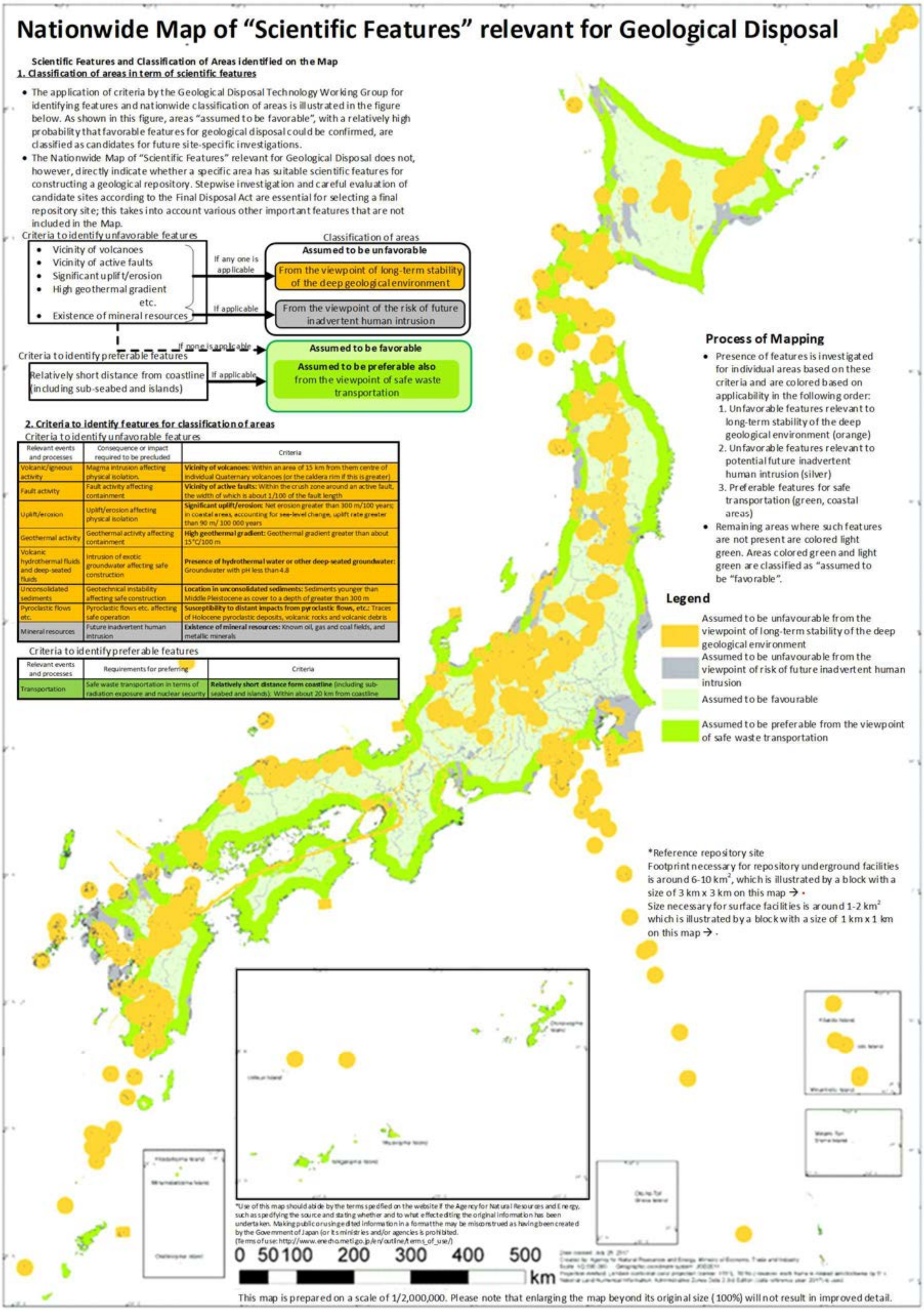

FIG. 2. Nationwide map of scientific features relevant for geological disposal. (Text on the map has been enhanced for visibility. The original can be viewed at https://www.numo.or.jp/en/jigyou/Explanation_material.pdf).

Paper ID#160
THORP – COMMERCIAL REPROCESSING AT SELLAFIELD

P. HALLINGTON
Sellafield Ltd
Seascale, Cumbria CA20 1PG
Email: phil.j.hallington@sellafield sites.com

Abstract

45 years since it was first conceived and after reprocessing over 9300 tonnes of fuel, THORP sheared its last fuel assembly on 9th November 2018. Providing a vital service to United Kingdom, European and international reactor operations, the facility will continue to store fuel for at least the next 50 years. This presentation will look back at some of the history, the economics and lessons learnt more than 24 years of successful operation and forward to a new future for the staff and facility.

1. INTRODUCTION

When British Nuclear Fuels Ltd announced plans to expand operations on the Sellafield site in 1975, one of the facilities mentioned was a new reprocessing plant to recycle thermal oxide fuel — THORP. The United Kingdom needed to build a plant to reprocess fuel from the British fleet of advanced gas cooled reactors, so BNFL developed with the ambitious plan to build a plant with more capacity than the United Kingdom needed and secure contracts with overseas customers. The economic case was a simple one: secure overseas contracts where the money was paid up front to build THORP (therefore reducing the burden on the United Kingdom Treasury) and build a larger plant than the United Kingdom needed (therefore spreading the operating cost of a plant which the United Kingdom needed to build). This would put United Kingdom in a strong commercial position to support the Global Nuclear Industry.

2. ORIGINS

When the plan for THORP was announced in 1975, it was not envisaged that it would be a further 19 years before the first fuel would be introduced into the shear cave. From the outset, there was tension. BNFL entered into a planning arena which was hostile to the project, involving the national press and international anti-nuclear groups which escalated into a public inquiry and debates in UK parliament. The 100-day Windscale inquiry heard evidence from all sides and at the end presiding inspector Mr Justice Parker recommended that the United Kingdom should reprocess thermal oxide fuel and that once built, THORP should be allowed to reprocess fuels from overseas. Objectors appealed to the Secretary of State for another parliamentary debate, ahead of the final decision. After this second debate permission was given to construct THORP — with conditions attached relating to safety and environmental considerations.

In the latter stages of commissioning, further legal challenges were mounted to prevent the plant being operated. These were successfully overcome, and THORP began operation with the first fuel, from Heysham station, being sheared in March 1994.

3. RESEARCH, DEVELOPMENT AND DESIGN

THORP was the third reprocessing facility to be built at Sellafield. The intended feedstock of oxide fuel required extensive flowsheet research and development programmes to provide the necessary technical information to support plant design activities.

Fundamental differences in the physical make-up of fuels to be reprocessed, required development of the highly reliable mechanical shearing equipment, batch dissolution in boiling nitric acid, and removal of fine particles from the active feed to the solvent extraction facilities.

The solvent extraction flowsheet was based on a PUREX-type process, modified however to provide for an early split between uranium and plutonium in the interests of improved utilisation of solvent, and reduced environmental impact downstream.

Two factors required these changes:

— The Plutonium content of the higher burnup oxide fuels;
— The need to process fuels of higher specific activity whilst reducing quantities of liquid effluent arisings.

The first factor led to development of pulsed columns for stages of the process where significant quantities of plutonium were present. The second factor resulted in the adoption of a 'salt-free' flowsheet.

Designing such an integrated facility brought many new challenges. The scale of the project, encompassing fuel receipt and storage and the main reprocessing facilities was 0.5 km long. To meet programme requirements, detailed design, construction and commissioning activities were carried out in parallel. Phased programmes continued from 1983 with the start of fuel receipt construction, until completion of main THORP construction in 1992.

Extensive use was made of the then latest computer aided design tools to produce integrated information on piping, vessels and cabling, which was linked to installation data to maximise productivity at the workface. This information was also used to support safety case development and commissioning testing, to bring the plant into operation.

4. THE PROCESS

THORP provided an integrated approach to reprocessing and recycling spent irradiated oxide fuels. Within the building envelope (Fig. 1) were facilities to receive and store spent fuels, mechanically shear fuel assemblies, produce fuel solution and separate uranium, plutonium and fission product wastes. In addition, uranium was purified into stable powder form ready for re-enrichment, and the plutonium was converted into oxide powder, safely packaged and stored ready for re-use.

The plant comprised:

— Receipt and Storage.
  - Flask unloading;
  - Fuel storage;
  - Transfer for reprocessing.
— Head End.
  - Fuel element verification;
  - Whole assembly shearing into large scale batch dissolvers with off-gas treatment;
  - Dissolver liquor clarification in solid bowl centrifuges;
  - Accountancy and buffer tanks.
— Chemical Separation.
  - Solvent extraction in pulsed columns for plutonium baring streams;
  - Solvent extraction in mixer-settler equipment for all other duties;
  - Transfer of highly active fission products for treatment on-site.
— Effluent Treatment.
  - Pre-treatment to remove oxide compounds;
  - Evaporation and nitric acid recovery;
  - Process off-gas treatment;
  - Low active effluent transfer systems.
— Conversion of Products.
  - Production of uranium trioxide powder in secure packaging and storage;
  - Production of plutonium oxide powder in secure packaging and storage.
— Central Services.
  - Chemical reagent preparation and supply;
  - Filtration and ventilation of process cells and facilities;

- Centralised decontamination and maintenance;
- Solid radioactive waste handling and transfer;
- Highly active analysis laboratories.

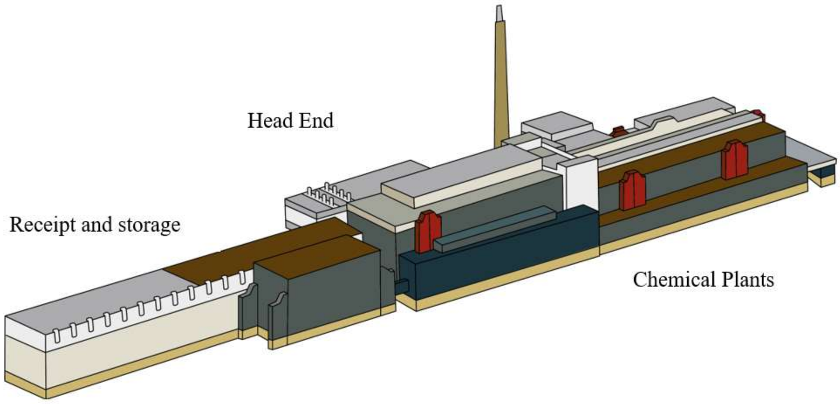

FIG. 1. Facility schematic.

FIG. 2. THORP 2018.

From the outset THORP was designed to operate to modern safety and environmental standards. This was achieved by incorporating many advanced features, to provide high reliability in operation and low maintenance. Extensive use was made of stainless steel within containment cells to provide for ease of future clean out and subsequent decommissioning.

Movement of liquids was carried out using power fluidic technology to avoid pumps and moving parts subject to wear and needing maintenance. A distributed control system, essentially a network of over 50 local processing stations linked to a central supervisory control computer system provided highly reliable process control, with high levels of redundancy built in.

5. SAFETY AND ENVIRONMENTAL ASPECTS

THORP was built to modern safety standards including seismic design criteria, and low levels of worker and public exposure to ionising radiation.

All reprocessing operations were conducted behind massive shielding, generally incorporating facilities for remote repair and maintenance. Secondary containment was provided in the form of in-cell stainless steel cladding, which also facilitates future decommissioning.

The original workforce radiation exposure design target was an individual exposure not exceeding 15 mSv per year. In practice, average exposure of the workforce has been around 0.18 mSv per year in 2018.

In terms of environmental impact, a range of modern effluent treatment facilities were commissioned at Sellafield in the 1980s. They were therefore available to support Thorp operations from the beginning. The 'salt-free' flowsheet and the collection of C-14 from the dissolution stage have further reduced the environmental impact of Thorp throughout its operational life.

6. OPERATIONAL EXPERIENCE

Over a period of almost 25 years of operation, over 9000 te(U) of oxide fuel has been successfully reprocessed. The initial operating period saw a steady increase in annual throughput towards ~900 te(U).

As was foreseen during the design stage, a range of operational issues have arisen. The conservatism and redundancy built into the plant allowed these challenges to be overcome and operations to continue – albeit in some cases at a lower throughput than design. Solutions to restore capacity were developed and tested, however due to changes in demand schedules for future processing they were never implemented.

Perhaps the most notable event occurred in 2005 when a significant quantity of dissolver product solution leaked into the stainless-steel cladding in-cell. The cause was later identified as weld failure in piping subject to different stress cycles resulting from a changed operating regime. No release of radioactivity occurred, all the material was recovered back into the process using the designed in provisions, and the plant was subsequently brought back into service. The solvent extraction plant has been very reliable throughout, with better than predicted flowsheet performance.

Experience with supporting downstream plants such as the facility encapsulating solid waste in cement grout have generally proven to be reliable – given their innate simplicity this is unsurprising.

During design, provision was made to capture and store products of solvent degradation that were anticipated to arise in operations. Experience has shown that the amounts arising are much lower than expected. Tank capacity will now be used to support post-operations clean-out.

Some plant items were expected to be replaced within an operating cycle. Items such as the shearing machine blades were remotely replaceable. Other equipment such as effluent evaporators were replaced at scheduled outages. Design provision to support this approach has been very successful.

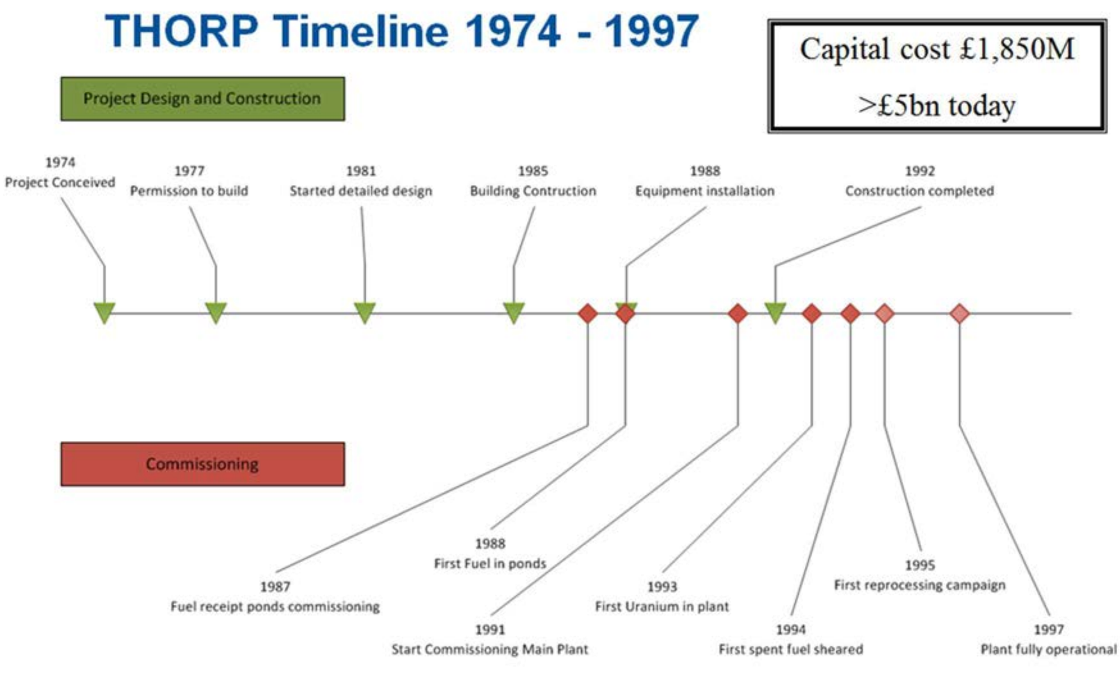

FIG. 3. Summary timeline to start of operations.

7. SAFEGUARDS

The need to meet the highest international standards of nuclear material accountancy and control were designed into THORP from the very beginning. Close liaison with Safeguards Authorities during design, construction and commissioning, has continued throughout the following ~25 years of operations. THORP deployed a very advanced (for the time) system of Near Real Time Material Accountancy (NRTMA) which was derived from the IAEA programme LASCAR. Independent data gathering from Key Measurement Points, incorporated throughout the whole process is a cornerstone of safeguards compliance. Equipment was installed to verify fuel element feed into the shearing equipment, based on advanced radiometric techniques. This has proven to be very reliable in-service. The automated system for transferring plutonium into secure and safeguarded storage has similarly met international expectations.

8. CLOSURE

The last batch of fuel was sheared into the Thorp dissolver on 9th November 2018, almost 25 years since the start of operations. This brought an end to commercial reprocessing at Sellafield, and the THORP facility has moved promptly into post-operational clean-out, making extensive use of the designed in provisions to support decommissioning which are a key feature of modern facilities at Sellafield.

9. THORP – INTO THE FUTURE

Some facilities in the building will still be used to support the next stages of decommissioning of the site. These are expected to take until at least 2032 to complete. In April 2019, enhanced cleanout procedures will begin. Using a systematic approach, different wash solutions will remove activity out of the plant making it more efficient to decommission when that stage arrives. This will also allow much of the equipment installed in THORP to go to low level waste routing for disposal, rather than medium or high level routes. This is in line with the site strategic ambition to significantly reduce the overall cost of managing the nuclear liability and hence bringing overall benefit to the United Kingdom.

During the final THORP Chemical Plants rundown, the vast majority of fissile material and chemicals will be removed from vessels and pipework. Residual radioactivity and chemicals will then be flushed out and the plant washed out. Further reduction of the radioactive and chemical inventory will follow, enabling the vessels and pipework to be removed and disposed of. As noted above, Thorp is also home to the medium active evaporator which supports the rest of the Site and is destined to do so until around 2032. The storage pond attached to the THORP reprocessing plant will continue in service for the coming decades for the long term storage of Advanced Gas-cooled Reactor (AGR) fuel. The fuel will be consolidated in other site facilities and transferred for long term storage prior to final disposal around ~2085.

To achieve this, the pond in Thorp will be re-equipped with new fuel storage racks to allow it to store all of the United Kingdom's un-reprocessed AGR fuel in safe long term storage pending final conditioning and ultimate disposal in a deep geological facility.

10. CONCLUSION

In the 1970s the UK sought to establish a new modern world-scale oxide fuel reprocessing plant, built to deliver safe secure reliable processing with a very small environmental impact

The Thorp plant has operated safely and securely for almost 25 years, completing commercial reprocessing business and treating over 9300 te(U), with very low environmental impact, in addition, the delivery of internationally significant nuclear material safeguards has been consistently achieved at industrial scales.

In all aspects, a very successful commercial project.

APPENDIX 1 - KEY DATES IN THORP 'LIFETIME'

Date	Event
1969	Oxide fuel reprocessing starts at Windscale in an early generation Magnox reprocessing facility which has been converted into a Head End facility capable of reprocessing oxide fuel. First 'modest' contracts signed to reprocess oxide fuel on a commercial basis.
1973	Original Head End plant which reprocessed oxide fuel is put out of action following a safety incident.
1975	BNFL announce plans for expansion of site operations – including Thorp BNFL contracts have been negotiated, to reprocess 1500 tonnes fuel for Japanese, German, Swiss, and Spanish utilities companies
1976	Government agrees to BNFL taking on reprocessing contracts from overseas, subject to return of wastes First parliamentary debate on Thorp Planning permission submitted for Thorp New international legislation on the exchange of nuclear materials between countries is introduced which requires the return of waste generated in reprocessing to the country where the spent fuel originated from.
1977	Secretary of State for the Environment "calls in" planning application. June - Public Inquiry launched. November - Public Inquiry closed
1978	Public Inquiry report published, allowing Thorp to be built
1981	Site clearance begins
1984	Major civil works start
1986	BNFL announces Thorp's order book is full after signing £1.6bn contracts with overseas customers and the English and Scottish electricity boards
1987	An analysis shows that Thorp will be capable of reprocessing 7000 tonnes of fuel in its first 10 years of operation – meaning there is an extra 1000 tonnes of capacity available to sell to foreign customers. They all take up the option.
1988	Thorp receipt and storage opens and receives first fuel
1989	Completion of main buildings and civil engineering contract Electrical and instrumentation installation started
1991	Completion of electrical and instrumentation installation
1992	All shear cave equipment installed and tested Timetable for revised Sellafield site discharge authorisation agreed with Regulators and Government.
1993	Third Parliamentary debate – covering the need for Thorp Summer - Second discharge authorisation consultation opened and "Trust Us" campaign launched October - Second discharge authorisation consultation closed December - Greenpeace attempt to gain judicial review and Secretary of State for the Environment announces the decision to go ahead with Thorp operations Government review of the viability of reprocessing allows Thorp to open
1994	First active shear of fuel – AGR from Heysham power station
1995	Official inauguration
1997	Plant fully operational
2012	Government announces Thorp reprocessing will cease in 2018
2018	The last fuel to be sheared in the plant took place in 2018 bringing to an end almost 25 years of operation.

Paper ID#25
SPENT NUCLEAR FUEL MANAGEMENT IN RUSSIA: STATUS AND FUTURE DEVELOPMENT

A.V. KHAPERSKAYA
State Corporation “ROSATOM”,
Moscow, the Russian Federation
Email: AVKhaperskaya@rosatom.ru

O.V. KRYUKOV
State Corporation “ROSATOM”,
Moscow, the Russian Federation
Email: OVKryukov@rosatom.ru

K.V. IVANOV
State Corporation “ROSATOM”,
Moscow, the Russian Federation
Email: KoVlIvanov@rosatom.ru

Abstract

At present, Russia's nuclear power industry continues its development and increases its contribution to the overall energy mix, which reached 18.9% in 2017. The basis of nuclear power generation is formed by LWR, in the same time Russia operates two industrial-size fast reactors — BN-600 and BN-800. It is expected that from the year 2030 there will be the large-scale implementation of fast neutron power reactors and the transition to a two-component nuclear system with unified fuel cycle, linking the needs of both existing thermal reactors and fast neutron reactors. Solving the problems associated with the accumulation of SNF and radioactive waste in this regard is becoming a priority.

As a basic approach to SNF management in Russia, the concept of its reprocessing with the nuclear materials recycling in a two-component nuclear power energy system (with thermal and fast neutron reactors) has been adopted. The main purposes are an efficient use of natural uranium resources, SNF non - accumulation, recycling nuclear materials, and reducing the radiotoxicity and volume of the generated radioactive waste.

Russia has many years of experience in safe management of spent nuclear fuel from power reactors including storage, reprocessing and recycling. The reprocessing plant RT-1 has been operating since 1977. To date, over 6000 tons of SNF have been reprocessed. At the same time, an integrated complex for SNF management is being created at the site of the Mining Chemical Combine, which includes: centralized water cooled (“wet”) SNF storage; centralized air-cooled (“dry”) SNF storage; a pilot-demonstration centre for the reprocessing of SNF based on innovative technologies; MOX fuel fabrication for fast neutron reactors (BN-800 type). An underground research laboratory will be set up here to develop the technologies for the HLW final isolation.

The recycling of reprocessed uranium (repU) is currently being fully implemented during the fabrication of fuel for thermal reactors. Separated plutonium from LWR SNF starts involving in NFC as a component of MOX fuel for FR (for starting loading and feeding during the first 10 years of operation of fast reactors). At the same time, the technology of multi-recycling in thermal reactors of plutonium and repU from LWR SNF is being developed (REMIX-concept).

To reduce radiotoxicity and the volume of ultimate wastes to be disposed of, HLW partitioning technologies are being developed with MA and heat-generating fission products recovering. Russia already has industrial experience in HLW partitioning.

The technology of MA transmutation is planned for studying using both solid-fuel fast reactors (like BN-800 type) and MSR.

1. INTRODUCTION

The Russian Federation has 35 operating nuclear power reactors (thermal reactors: VVER-1000/1200: 15 units, RBMK-1000: 10 units, VVER-440: 5 units, EGP-6: 3 units, Fast reactors: BN-600 - 1 unit, BN-800 - 1 unit); 6 units VVER-1200 type. The first unit of Floating NPP (SMR) are under commissioning. 7 units of nuclear power reactors different types are in various stages of decommissioning. The planned layout of future NPPs at the territory of the Russian Federation has been set out by the Government Order of the Russian Federation no. 1634-r of August 1, 2016. List of nuclear power plants scheduled for construction until 2030 includes 11 new power units.

As a basic approach to SNF management in Russia, the concept of its reprocessing with the nuclear materials recycling in a two-component nuclear power energy system (with thermal and fast neutron reactors) has been adopted for the purpose of efficient use of natural uranium resources, SNF non-accumulation, recycling nuclear materials, and reducing the radiotoxicity and volume of the generated radioactive waste.

2. SNF MANAGEMENT IN RUSSIA: STATUS OF INFRASTRUCTURE

The core principle of the state policy of the Russian Federation in the field of SNF management involves SNF reprocessing to prevent SNF accumulation, to recycle the nuclear material with recovering the remaining energy resource of the SNF and reducing the uranium demand, and to ensure the environment-friendly management of RW (FP& MA). The task of ensuring the safe management of RW is considered to be, on the one hand, a key element of the national security and safety, and, on the other hand, an essential precondition for present and future use of atomic energy.

The Energy Strategy of the Russian Federation Until 2030, approved by the Government of the Russian Federation, provides for the following efforts in the field of Nuclear Fuel Cycle and Nuclear Power:

— Upgrading NPP capacities with thermal-neutron reactors;
— Construction of experimental and commercial power plants with fast breeder reactors;
— Implementation of a closed nuclear fuel cycle involving new technologies and new enterprises;
— Development of advanced RW treatment methods, technologies, and ensuring closed nuclear fuel cycle when the rate of waste accumulation is equivalent to the rate of waste disposal.

Centralized SNF management in Russia is providing at two sites: PA Mayak and Mining & Chemical Combine (MCC).

2.1. RT-1 plant at "PA Mayak"

Industrial-scale SNF reprocessing is performed at RT-1. Plant RT-1 at "PA Mayak" has been operating since 1977. Now, about 6000 tonnes SNF have been processed. Processed SNF inventory includes almost all existing uranium and plutonium compositions and cover all the FAs dimensions. The design capacity is 400 tonnes per year. At present, the SNF of VVER-440, BN-600, SNF, RR SNF, defect fuel of RBMK (which cannot be accommodate to the dry storage) is reprocessed at the RT-1 plant, the reprocessing of VVER-1000 SND has been started in 2016. Necessary infrastructure is being set to enable AMB and EGP-6 SNF reprocessing. Mixed oxide uranium-plutonium (MOX) irradiated nuclear fuel (SNF) of FN-600 reactor was reprocessed at RT-1 plant in 2012 and 2014 [1]. Reprocessing is based on PUREX-process (“modified PUREX”) involving the extraction of recycled uranium and plutonium as target reprocessing products with a possibility of extracting neptunium, as well as a broad range of other isotopes (Cs-137, Kr-85, Am-241, Pu-238, Sr-90, Pm-147). A lot of attention to environment issues was paid in recent years for legacy sites rehabilitation, open RW pools were decommissioning, new complex of cementation and new vitrification furnace was put into operation. Aluminophosphate glass is used for the vitrification of the HLW after the reprocessing. Borosilicate glass will be also used in near future. The first in the world semi-industrial facility for partitioning of high-level wastes was put in operation at RT-1 in August 1996. SNF reprocessing is accompanied with a production of radioactive waste which are subjected to treatment. Current practice for ILW and HLW management from SNF reprocessing at RT-1 plant involves HLW vitrification in EP-500 ceramic melter with design capacity 500 l of concentrated HLW per hour. Aluminophosphate matrix of the radioactive glass is produced using direct evaporation-calcination-vitrification technology. Vitrified wastes are placed in steel canister and are stored in a dry vault-type storage facility.

2.2. The integrated complex for SNF management at the mining and chemical combine

At the same time, the integrated complex for SNF management is being created at the site of the Mining and Chemical Combine, which includes: centralized water cooled ("wet") SNF storage; centralized air-cooled ("dry") SNF storage; a pilot-demonstration centre for the reprocessing of SNF based on innovative technologies; MOX fuel fabrication for fast neutron reactors (BN-800 type). An underground research laboratory will be set up here to develop the technologies for the HLW final isolation.

2.2.1. VVER-1000 SNF wet storage.

The storage represents the facility with the high level of safety and seismic stability. Presently the newly arrived SNF is placed in "wet" storage facility, and the aged fuel is transferring from the "wet" storage facility to the "dry" one.

2.2.2. VVER-1000, RBMK-1000 SNF dry storage facilities

The storing complex for RBMK-1000 and VVER-1000 SNF, in its full-scale development, composed of three buildings, dry vault-type storage facilities, were put into operation and were successfully operated since 2011. Passive safety concepts are applied in SNF "dry" storing technology.

2.2.3. MOX-fuel fabrication for fast neutron reactors

Presently the facility is in operation and produces the fuel for reactor plant BN-800 (Beloyarsk NPP). The production provides a possibility of FAs fabrication with the separated Pu from power reactors SNF.

2.2.4. The Pilot Demonstration Centre on SNF reprocessing based on innovative technologies

PDC is an integral component of the integrated complex for SNF management at MCC. PDC is designed to reprocessing LWR SNF (VVER-1000 type, RBMK, PWR and BWR — there is a possibility for reprocessing). The key goal of the PDC innovation technologies development is to achieve ecologically acceptance and economically efficiency of the reprocessing technologies. The PDC is constructing in two stages. In 2016, license was granted to operate first start-up complex of PDC. This unit involves hot research cells, analytical facilities, as well as other necessary infrastructure. R&D programme aimed at elaborating innovative SNF reprocessing technologies has been launched in 2016. The purpose is to confirm the designed parameters of the new technological scheme, further improvement of new technologies for reprocessing of SNF; development of HLW partitioning technologies for reducing radio toxicity of ultimate disposal waste.

Construction of the second PDC section with a design capacity of 250 tonnes of SNF per year is underway. It is scheduled to be commissioned in 2021. The reprocessing technologies were developed (based on the Simplified PUREX process) to provide the absence of liquid radioactive wastes (effluents) discharge. The main products of PDC - mixed oxides of plutonium, neptunium and uranium for manufacturing of the fast reactor fuel, as well as repU. PDC is also ready to deliver fuel product for REMIX. HLW are vitrified in borosilicate glass for further ultimate disposal to vitrification in borosilicate glass.

3. RECYCLING TECHNOLOGIES DEVELOPMENT

Regenerated nuclear materials (repU and Pu) have been traditionally used in Russia separately. RepU is reusing in Russian commercial nuclear reactors (RBMK type, BN VVER–440 VVER–1000) since 1996. At present, Russian fabrication plant MSZ has a license for reprocessing nuclear materials based on repU with ^{232}U content up to $5 \cdot 10^{-7}$ %.

Separated plutonium from LWR SNF starts involving in nuclear fuel cycle as a component of MOX fuel for fast reactors (for starting loading and feeding during the first 10 years of operation of fast reactors). The concept of two-component nuclear energy system has approved in Russian Federation, including both reactor types (VVER and BN). The transition period will include reusing in Russia of reprocessed nuclear materials as mixed fuel for LWRs and FR.

The technology of multi-recycling of plutonium and RepU from LWR SNF in the form of fuel for the existing and future fleet of thermal reactors (VVER-1000 type) is being developed in Russia (REMIX-concept).

REMIX fuel is the mixture of U and Pu from LWR SNF reprocessing, with the addition of enriched uranium (natural or reprocessed U). REMIX fuel enables multiple recycling of the full quantity of U and Pu from spent fuel, with the 100% core charge and saving of natural uranium in each cycle. Compensation accumulated even isotopes of U and Pu by the natural uranium feeding allows performing up to 7 recycles. The main advantage of REMIX this technology is that U-Pu mix can be incorporated into the reactor fuel enabling multiple recycling of uranium and plutonium in thermal reactors. REMIX composition does not require reactor modification for the thermal neutron reactor like VVER (or PWR, BWR) and replacing fuel from natural U on REMIX fuel does not require capital expenditures from NPP operator.

State Corporation ROSATOM develops a programme for REMIX fuel implementation. In the framework of the programme three experimental REMIX fuel assemblies; (FA) containing 18 REMIX-fuel elements have been manufactured. Since 2016, they are being irradiated at Balakovo NPP. In parallel, ampules for FA irradiation in MIR research reactor and post-irradiation investigations were manufactured and are under irradiation (up to 2020)– some of ampoules have already been removed and are being investigated. In 2018 there was the start of the safety case development programme for REMIX fuel use in VVER-1000 and VVER-1200 reactors. The programme includes the development and validation of computer codes for nuclear and radiation safety demonstration. There are plans for the industrial facility for REMIX-fuel fabrication construction. In 2018, the development of investment justification such facility was initiated and will be finished in 2019.

4. HLW PARTITIONING TECHNOLOGIES DEVELOPMENT

To reduce radiotoxicity and the volume of ultimate wastes to be disposed of, HLW partitioning technologies are being developing with MA and heat-generating fission products recovering. Russia already has industrial experience in HLW partitioning. Since 1996, the Mayak (RT-1) plant has operated a pilot plant for HLW partitioning. During the operation more than 1200 m^3 of HLW was processed with Cs-Sr recovering. At present, a process using N, N, N ', N'-dioctyl diamide diglycolic acid (TODGA) as an extractant in a polar diluent meta-nitrobenzotrifluoride [1], followed by chromatographic separation of americium and curium has developed in the Russian Federation. This technology includes both liquid extraction and sorption processes. The "hot" dynamic tests of the technology for recovering Am and Cm from real HLW by the TODGA system were carried out. The recovery of americium is more than 99.9%. Technology for the separation of americium and curium was demonstrated in the pilot plant at Mayak PA [2].

The programme for partitioning HLW technologies development includes:

- Maturing of HLW partitioning technology (with Am, Cm, RE, Cs-Sr recovering from HLLW and its separation) including modernization partitioning facility at Mayak plant,
- Developing and deployment facility of HLW partitioning facility at the MCC,
- Developing the technologies of Am, Cm oxides and mixed U-TPE oxides precipitation,
- Developing and maturing the technologies for MA-bearing fuel fabrication, fabrication, irradiation , PIE, recycling experimental Am- and Np-bearing fuel
- Complex database for fuel characteristics and codes for MA recycling.

5. THE TECHNOLOGIES FOR MA TRANSMUTATION

The technology of MA transmutation in Russia is planned for studying using both solid-fuel fast reactors (like BN-800 type) and MSR.

The following scenarios are considered when analysing the concept of MA management in fast reactors:

- Homogeneous transmutation of MA in the fuel;
- Heterogeneous transmutation of MA in special assemblies.

The necessary efficiency can be achieved both in the homogeneous and in the heterogeneous approaches.

Currently, both schemes are deeply researched within the framework of R&D for the "Breakthrough" project [3].

The programme of development of the MA transmutation technologies in FR includes:

— Justification of neutron-physical characteristics of core with MA and efficiency of transmutation of MA in FR, development of requirements for fuel with of minor actinides (including the experiments with MA- bearing FAs in BFS, in BOR-60 and BN-800);
— Development of the technologies of Np homogeneous recycling (design, fabrication, irradiation and post-irradiation examination of mixed oxide and nitride uranium-plutonium fuel with Np in amounts of 0.1% to 1%);
— Development of the technologies of Am recycling homogeneous (with content of Am 0.4–1.2%)- mixed nitride and oxide fuel, uranium nitride and oxide fuel; heterogeneous recycling (with a content of Am up to 10–12%)- mixed nitride and oxide fuel, uranium nitride and oxide fuel (design, fabrication, irradiation and post-irradiation examination). The result — the option with optimal performance of Am recycling in FR fuel cycle.

As an alternative option for MA burning, Russia develops the approach of burning MA in MSR [3].

The advantages of MSR as a TRU burner from SNF reprocessing are primarily due to the lack of the need to manufacture a fuel pellet and the possibility of widely varying the content of long-lived actinides in fuel salt without core modification.

The construction of a large power MSR is proposed to be preceded by the construction of 5–10 MWt Demo MSR unit to demonstrate the control of the reactor and fuel salt management with its volatile and fission products with different TRU loadings for startup, transition to equilibrium, drain-out, shut down etc. The development of the proposed technology on an industrial scale will certainly require solving of a number of technical tasks; however, there are no deadlock problems on this path.

6. CONCLUSION

Russia already operates nuclear system with thermal and fast neutron reactors and developing the new innovative technologies and infrastructure for future sustainable development with U-Pu multirecycling in thermal and fast reactors and reduction of radiotoxicity of radwaste to be disposed of. RT-1 (reprocessing facility in Mayak plant) is the Russian pioneer facility for power reactor reprocessing, with possibility of reprocessing almost all existing types of SNF, we have full scale recycle reprocessed uranium and start using Pu in fast rector fuel. The new complex of NFC facilities with new technologies of reprocessing and recycling as integrated plant is under creation in Krasnoyarsk area. There are already exist a complex of SNF storage facilities, industrial plant for MOX-fuel fabrication, and SNF reprocessing being deployed. An underground research laboratory supporting the R&D programme for RW deep geological disposal is also being constructed there. These prospects also suggest REMIX-fuel fabrication facility and, in a longer term perspective, operation of molten salt reactors to burn minor-actinides. Introduction of MA burning in FR and/or MSR into the Russian nuclear fuel cycle will allow solving the problem of utilization of long-lived actinides after SNF reprocessing.

NOMENCLATURE

PDC	Pilot demonstration centre
HLW	High level waste
MCC	Mining and Chemical Combine
VVER	Pressurized water reactor of Russian design
REMIX	Fuel for technology of multi-recycling of plutonium and repU in LWR
MSR	Molten-salt reactor

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3.2. TRACK 2 – SPENT FUEL AND HIGH LEVEL WASTE STORAGE AND SUBSEQUENT TRANSPORTABILITY

Overview prepared by T. Saegusa (Japan), J. Ljungberg (Sweden) and T. Tate (United States of America), **Track Leaders**

The spent fuel and high level waste storage and subsequent transportability track highlighted the current long term and interim storage activities associated with both wet storage and dry storage systems. Discussions were conducted regarding the implementation of ageing management programs to ensure fuel integrity and identify degradation mechanisms. Efforts to ensure the safety and security of spent fuel and high level waste were discussed and specific features that support safety and security in wet and dry storage applications were included. Furthermore, discussions on data collection and testing activities to improve the characterization of spent fuel and high level waste to improve and support storage and transportability were conducted.

Any component and material in spent fuel storage systems will degrade eventually. Therefore, Ageing Management Programmes (papers ID#22, ID#60, ID#103) with monitoring and inspection (papers ID#27, ID#104, ID#162) are essential for long term storage based on degradation mechanisms (papers ID#22, ID#88, ID#122, ID#134), integrating operating experience and lessons learned (paper ID#182). In this context, valuable presentations and discussions were made. Some highlights extracted from the respective papers are mentioned below:

Paper ID#60 The challenge of obsolescence or technology ageing is an often-unforeseen effect on various systems. Some more advanced water treatment systems would have been good if they had been introduced when the facility was constructed.

Paper ID#27 Described a new way of detecting leaks by measuring temperatures, independent of leakage position or size. The pressure decrease has to be high enough to allow the measurement. What needs to happen inside the container is a pressure decrease increment big enough for measurement. Measurement of the temperature on the top and the bottom of the canister is much easier than using leak detection systems.

Paper ID#104 There seems to be no detrimental corrosion effects ongoing for the Boral systems.

Paper ID#182 In the Fukushima accident the spent fuel pools and the dry storage systems were stressed. Although the spent fuel in pools or in dry storage was not damaged, the author highlighted the fact that Japan has chosen dry storage due to its passive safety which does not require electricity to sustain their function.

Other presentations include the following:

Paper ID#122 Perspective on thermal creep for dry storage and transportation applications;

Paper ID#147 Spent nuclear fuel storage: concepts and safety issues;

Paper ID#164 Thermal modelling round robin of the high burnup demonstration cask;

Paper ID#173 The pressurized water reactor (PWR) spent fuel dry storage project experience in China.

Session 2.1: Spent Fuel and High Level Waste storage and subsequent transportability: Ageing management of storage systems (wet & dry) (part 1)

Session Chairs: M. Lloret (Spain) and H. Takeda (Japan)

Session 2.1 comprised of six presentations, one from Canada, one from China, one from Sweden, one from United Kingdom, one from United States of America and one from Germany.

- **Paper ID#22 by A. Barry (Canada)** presented two long term storage experiments. Canadian Nuclear Laboratories (CNL) has conducted two long term fuel storage experiments since the 1970's for both wet and dry conditions. Wet storage experiment results indicated that spent fuel can be safely stored under water for more than 50 years and that intact spent fuel can be stored in dry or moisture-saturated air at up to 150ºC for over 15 years. The same storage conditions result in oxidation of defected fuel and fission product release, but no bulk fuel oxidation to U_3O_8 or loss of fuel integrity occurred. Future examinations are required for evaluation in longer time frame.
- **Paper ID#173 by B. Shangguan (China)** presented spent fuel interim storage project experience of the author in China, including implementation strategy, product selection, bidding, engineering, and construction. According to the Chinese project experience, the domestic design and manufacturing provides a significant price reduction. An analysis of the risks has been performed and the R&D route has been defined: requirement definitions, design, prototype manufacture, test and qualifications, and manufacture supply. Clear responsibilities for all parties in the contract, and accurate definition of the boundary of intellectual property were the key points to the success of the project. Their canister design and concrete silo design were introduced.
- **Paper ID#60 *(Invited)* by M. Nyström (Sweden)** presented about implementing ageing management programme in the Swedish Central Interim Storage Facility for Spent Nuclear Fuel (CLAB), owned by SKB. At CLAB, approximately 6700 tonnes of SF (35 000 fuel elements) are currently stored with a residual power of approximately 8.3 MW. The process of ageing management at CLAB consists of four main parts: scoping and screening, ageing management analysis, system analysis, reporting and follow-up. Several lessons were learned in the development of the ageing program, for example the importance of good communication with the supervisory authority, or the necessity of an adequate document management system
- **Paper ID#88 by A. Ledger (United Kingdom)** presented the strategy adopted by Sellafield Ltd for management of the remaining Advanced Gas-cooled Reactor (AGR) spent fuel from EDF reactors, after the closure of THORP reprocessing plant. Spent fuel will be wet stored in existing facilities for the interim period until a disposal facility becomes available, extending fuel storage time from the current 5 years (for buffer storage pending reprocessing) to 80 years. The author studied the fuel integrity and the pond integrity. For the fuel integrity, the wall thickness loss of fuel cladding during interim storage was investigated for both LWR and AGR. Studies regarding the integrity of fuel and its resistance to corrosion have been completed, and fuel is capable of being stored for up to 80 years.
- **Paper ID#104 by H. Akkurt (USA)** presented the Industrywide Learning Ageing Management Program (i-LAMP) proposed by the Electric Power Research Institute, Inc. (EPRI) to the industry for monitoring neutron absorber materials (NAMs) used in spent fuel pools (SFPs). As a part of the program, SFP water chemistry and coupon analysis are being collected; data collection and analysis will be completed by the end

of 2019. Some guidelines will be published by EPRI as recommendations. EPRI will host the SFP water chemistry and coupon databases, and access will be limited to EPRI members.

- **Paper ID#115 by M. Schwerdtfeger (Germany)** presented the BfE's research activities that should be performed in Germany due to the longer dry storage periods until final disposal. As some of the cask components can be replaced, during the lifetime of the facility, the research goal is to determine the timescale beyond the initial plan of 40-year storage to duration in which the present concept is considered safe from a licensing perspective. This is mainly a research in ageing and degradation mechanisms. Additionally, research focused on repair, and maintenance are also very important, as the inspection, repair, maintenance and / or exchange of components or equipment requires quality staff and facilities as well as components.

Session 2.2: Spent Fuel and High Level Waste storage and subsequent transportability: Ageing management of storage systems (wet & dry) (part 2)

Session Chairs: H. Akkurt (United States of America) and M. Nyström (Sweden)

Session 2.2 comprised of five presentations, two from the USA, two from Japan and one from France.

- **Paper ID#27 by H. Takeda (Japan)** presented development of helium leak detection methods for canisters. The speaker highlighted the fact that current concrete casks do not have monitoring systems despite the need to detect any potential leaks to confirm the confinement. The speaker presented potential approaches for leak detection and highlighted that with the proposed approach leaks can be detected.
- **Paper ID#103 by P. Narayanan (USA)** presented ageing management of dry storage systems in centralized interim storage facilities. The speaker presented an overview of dry storage systems, including a horizontal canister system and discussed ageing management requirement for on-site as well as for the proposed centralized storage system. The speaker highlighted that centralized storage offers benefits over on site independent spent fuel storage installation (ISFSI) due to site location and innovative system designs.
- **Paper ID#162 by J. Renshaw (USA)** presented an overview of dry storage inspection, mitigation, and repair activities. The speaker highlighted that due to the collaborative efforts spanning across many organizations, development of several inspection methods was accomplished in an expedited manner. The developed inspection systems were tested not only using mock-ups but also using fully loaded canisters at different sites. The speaker also discussed ongoing collaborative activities toward development of mitigation and repair approaches.
- **Paper ID#182 *(Invited)* by S. Kaminishi (Japan)** provided an update on Fukushima Daichi nuclear accident and discussed the lessons learned from the accident with respect to spent fuel storage. The spent fuel pools (SFPs) for Units 1–4 were affected by tsunami and loss of power; subsequently cooling functions and heat removal were lost. Later analysis showed that fuel was not damaged under these conditions. The seawater, sand and rubble that entered in the dry storage building did not affect the fuel integrity. Removal of fuel from Units 1–3 SFPs is ongoing.
- **Paper ID#134 by C. Roussel (France)** presented the borosilicate glass HLW stability during long term interim storage. The speaker presented an overview of the recycling strategy in France and glass canister characteristics. It was highlighted that although the

storage facility concept is a simple passive system (based on convection), the glass temperature remains below 510ºC and therefore, shows good thermal stability during interim storage.

Session 2.3: Spent Fuel and High Level Waste storage and subsequent transportability: Safety and security of dry storage

Session Chairs: K. Agarwal (India) and A. Presta (France)

Session 2.3 comprised of six presentations, one from Ukraine, one from Iran, one from France, one from USA, one from Egypt and one from Japan.

- **Paper ID#3 by S. Alyokhina (Ukraine)** presented a scientific basis and a brief result of thermal analysis of dry storage spent nuclear fuel of Zaporishka NPP. The analysis compares the results from varying wind speed and directions along with varying ambient temperatures to estimate the surface temperature of the container and matches with experimental measurements.
- **Paper ID#7 by A.M. Taherian (Iran)** discussed design of casks for transportation and concrete cask module for long term storage of VVER 1000 spent nuclear fuel from Busher NPP. The presentation gave details on increasing the storage capacity of dry storage cask after carrying out radiation shielding analysis and criticality calculations.
- **Paper ID#147 *(Invited)* by F. Ledroit (France)** presented spent fuel storage concepts comparing wet and dry storage for enriched recycled uranium (ERU), mixed oxide (MOX) and enrichment natural uranium (ENU) spent fuel. The conducted analysis, comparing dry casks fully loaded with MOX fuels with full load of ERU or ENU fuels concluded that MOX fuel would take several decades longer to bring down the heat load of 2 KW per assembly in comparison to ENU or ERU spent fuel. Actual fuel inventory including fuel types (ENU, ERU, MOX) affects the choice of storage technology. It is important for countries using or planning to use partial MOX fuel core management in reactors.
- **Paper ID#164 by A. Csontos (USA)** presented a round robin test carried out on thermal modelling for high burnup spent fuel storage in a demonstration cask using three different techniques by different agencies for predicting surface temperatures and cladding temperatures.
- **Paper #83 by Z.F. Hassan Akl (Egypt)** presented regulatory aspects for safety and security of upcoming spent fuel storage facility outlining the mechanisms and procedures built in their regulation system to eliminate any unauthorized intrusion into the facility.
- **Paper #190 *(Invited)* by T. Narita (Japan)** presented the regulatory efforts on the current spent fuel management in Japan, based on TEPCO Fukushima Daiichi NPP Accident in 2011. The main topic was that the regulatory body has revised its regulation on the Dual-Purpose Cask (DPC) to streamline applications and promote the usage of DPC. The amendment of the regulation was from both technical and institutional viewpoints. DPC can be utilized at any NPP site for storage and for transport.

Session 2.4: Spent Fuel and High Level Waste storage and subsequent transportability: Behaviour & management of spent fuel (wet and dry)

Session Chairs: M. Schwerdtfeger (Germany) and S. Alyokhina (Ukraine)

Session 2.4 comprised of six presentations, one from European Commission, one from South Korea, one from USA, one from Germany, one from Spain and one from Hungary.

- **Paper ID#51 by D. Papaioannou (European Commission)** described the experimental facility used for the mechanical loading tests on irradiated LWR fuel. Two types of tests were conducted - hot and cold - and their results were presented. Experimental tests have been simulated for both static and transient with ANSYS code.
- **Paper ID#86 by D. Kook (Korea)** described the situation of the spent fuel management in South Korea. The author presented and discussed the experimental data: the rod internal pressure for cladding, the fuel temperature evaluation, cladding creep, the hydride reorientation in cladding and delayed-hydride cracking. Integrity evaluation platform and fuel assembly hardware were also presented in the work.
- **Paper ID#122 by A.J. Machiels (USA)** discussed thermal creep mechanisms in the low and high stress regions and their dependence to applied stress. It was highlighted that low and high stress are governed by different mechanisms.
- **Paper ID#114 by K. Linnemann (Germany)** presented on the management of damaged fuel. Failure mechanism and systems for encapsulation were described. The paper contained results of criticality assessment and deformation under thermal influence; the structure of closure system and quality assurance were shown and explained.
- **Paper ID#169 by M. Lloret (Spain)** presented PWR NPP Integral Model Fuel Assembly classification and the methodology for Cladding Hoop Stress Calculation. Assembly repairs were discussed and support to cask vendors for calculations was proposed.
- **Paper ID#47** ***(Young Generation Challenge Winner)*** **by B. Ficker (Hungary)** described structure of the SNF storage facility in Hungary and approaches to enhance its capacity, highlighted challenges and possible solutions. Studies were accompanied by numerical simulations and calculation of criticality and thermal profiles of stored fuel inventory.

Paper ID#60
IMPLEMENTING AGEING MANAGEMENT PROGRAMME IN INTERIM WET STORAGE

M. NYSTRÖM
Swedish Nuclear Fuel and Waste Management Co
Oskarshamn, Sweden
Email: mattias.nystrom@skb.se

Abstract

CLAB is a facility, own by Swedish Nuclear Fuel and Waste Management Co (SKB), for wet intermediate storage of spent nuclear fuel pending deposit in final repository. At CLAB 6700 tonnes of nuclear fuel are currently stored with a residual power of about 8.3 MW. Requirements regarding ageing management programs at nuclear facilities were introduced in 2006. Two attempts to introduce ageing management programs were made between 2006 and 2013 but failed. In 2013, SKB received an injunction to implement an ageing management programme for CLAB.

A project group was appointed to produce an appropriate ageing management programme for the facility. The programme was developed with guidance from the IAEA safety guide No NS-G-2.12.

The strategy was to involve the line organization early in the project so that the people who would manage the programme were involved in the development of management systems, analyses and proposals for measures. This made the handover of the project to the line organization quite simple and straight forward.

The facility consists of approximately 160 systems, of which about 96 are included in the ageing program. Only systems that are important for radiation safety are currently included in the program.

After all systems were analysed from an ageing perspective, 546 new measures, were identified that needed to be implemented to have control over the facility's ageing. During the execution of the measures, several unexpected discoveries have been made.

The result of the work in the programme has shown that the plant's status with regard to physical ageing is good. Technological ageing (obsolescence) is a bigger challenge.

Several lessons were learned in the development of the ageing program, for example the importance of good communication with the supervisory authority. Another lesson is the importance to set the right level of analysis that otherwise risks becoming ineffective.

1. INTRODUCTION

CLAB (Central Interim Storage Facility for Spent Nuclear Fuel) is a facility owned and operated by Swedish Nuclear Fuel and Waste Management Co (SKB). CLAB is a facility for wet intermediate storage of spent nuclear fuel pending deposit in final repository. CLAB was commissioned in 1985 and consists of an on-ground part with unloading equipment and auxiliary systems and an underground part with storage pools. At CLAB, approximately 6700 tonnes of nuclear fuel (≈ 35 000 fuel elements) are currently stored with a residual power of about 8.3 MW. At the plant, all spent nuclear fuel is stored from Sweden's nuclear power program. The plant also stores other components from the nuclear power plant's primary system, such as control rods and neutron detectors. The facility is planned to be in operation until the middle of the 2060s when the last fuel elements, according to the planning, will be deposited in the upcoming final repository.

Requirements regarding ageing programs at nuclear facilities were introduced in Sweden through authority regulations in 2006 [1]. Two attempts to introduce ageing management programs were made between 2006 and 2013. In 2013, the supervisory authority carried out monitoring with regard to the ageing management at CLAB. The authority then assessed the programme as substandard and instructed SKB to develop an appropriate ageing programme through an injunction.

Contributing factor to the failure was related to a lack of competence about what an ageing management programme consists of. There was also a lack of prioritization which meant that sufficient resources were not assigned. The lack of knowledge meant that the organization did not understand how an ageing management programme should be built and what activities should be included. As a result, there was a programme on the paper, but there was no one within the organization that could distinguish the maintenance programme from the ageing management program. There was also no appointed person to be in charge of the ageing program.

In a first attempt to develop a program, in 2006, a consultant was assigned with the task to go through the existing maintenance programme and make a gap analysis. The gap analysis should show which additional measures that were required to handle ageing equipment with significance to radiation safety. The result was a list that showed a few gaps. The list became CLAB's first "ageing management program". No management or development of the work was done at this time, the list was just archived in the document management system.

As the organization could not distinguish the maintenance programme from the ageing programme or explain what the programme was, a new attempt was made in 2010. At this time, a free interpretation of an authority document [2] was made, the result was that the ageing programme would only cover the plant parts that were intended to keep the entire plant's lifetime. This means that the ageing programme would cover the control programme for structures (rock and concrete structures). The ageing programme was thus more limited than before. In practice, the approach did not differ from the ordinary preventive maintenance programme for structures.

In 2013, the supervisory authority carried out monitoring with focus on the ageing programme at CLAB. CLAB then received sharp criticism for the methodology and working methods regarding ageing programs. SKB was given an injunction to produce an appropriate ageing management programme for CLAB. This was the start for SKB to handle the issue seriously.

2. IMPLEMENTING OF THE AGEING PROGRAMME AT CLAB

When CLAB was given an injunction to produce an appropriate ageing programme in 2013, the maintenance manager was given a clear responsibility to rectify the injunction. It was found early that the effort would be substantial. When the responsible manager analysed the whole picture and what measures were required to rectify the injunction, a number of conditions emerged which were considered important for coping with the challenge. The most important conditions were considered to be:

— The work must be carried out as a project;
— The project must have a management group;
— The project must be staffed with in-house staff;
— The resources must work full time in the project;
— The resources should be hand-picked by the project manager;
— It must be stated that the company management prioritize the project.

The conditions were discussed with the management, which accepted them. The maintenance manager at CLAB was appointed project manager.

2.1. Project implementation when developing the ageing program

2.1.1. Manning the project

At the start of the project the project crew consisted of:

— 1 project manager;
— 1 senior operations engineer;
— 1 senior shift leader;
— 1 senior maintenance engineer;
— 1 maintenance engineer who was designated to manage the ageing programme in the line organisation.

All four project participants were handpicked by the project manager with the aim of getting a functional and creative project group. Participants were selected because of their good technical know-how, their skills and ability to deliver, as well as their different personal qualities.

The project group was placed at the facility in new offices in order to release their regular work for the benefit of the project.

All participants in the project's core group worked full time in the project. In cases where replacements were required for their regular services, this was added. For example, an acting maintenance manager took over responsibility for the maintenance at the facility during the time the project continued.

As the project progressed, additional staff was involved, including full time project participants and line staff for specific tasks. The strategy was to involve the line staff who would perform the tasks when the ageing programme was handed over to the regular organization.

Examples of competencies involved in the project were maintenance personnel, operating staff, chemistry personnel and fuel and material specialists.

2.1.2. The start-up phase of the project

The start-up phase of the project focused on analyzing and understanding the problem, an important prerequisite for success was to increase the project group's competence in the task.

The start-up phase can be roughly described with the following stages:

- Skills development;
- Analysis of injunction;
- Action plan.

Skills development was conducted as a separate study of the available documentation relating to ageing management, with subsequent workshops where documents were interpreted and discussed. The documents used came from IAEA, WANO, and the authority. The next step was to go through routines, instructions and activities that already were developed for the existing ageing program.

A definition of "ageing management program" was stated. This activity required a lot of discussions and time, but it was well-invested hours that provided good support during the rest of the project.

After the analysis of the injunction and the subdocument it became clear why it had been issued. The facility's ageing programme lacked, in principle, all the criteria required to be systematic, traceable and effective.

An action plane with time frame was drawn up and it was clear that it would be impossible to produce a ready-made ageing programme in the time required by authority. SKB requested more time, but this was denied.

2.1.3. Development of management system for the ageing program

It was decided early on that available models should be used. It was decided that CLAB's ageing programme should be based on the IAEA's safety guide No NS-G-2.12, Ageing Management for Nuclear Power Plants [3].

This stage of the project largely involved drawing process schedules and flow charts of the various activities. Initially, a lot of time was spent drawing the overall process for the entire ageing program. Based on this, flow charts were then made for the various activities within the program. By using the flow chart, it was found that five governing documents were required for the ageing program. The flow charts were worked out and establish in the organization before the governing documents were produced.

2.2. Handing over the project to the line organization

Handover to line organization were undramatic because the line was already committed in the project and had been involved in developing the method and analyses. The handover consisted mainly of the fact that the project group was dissolved. The personnel responsible for completing and managing the programme were already familiar with the tasks as they had participated in the project.

The strategy for staffing the project with in-house staff and performing the work internally proved to work as intended. Since the staff responsible for carrying out the practical measures themselves had been included in the analyses, there was an understanding and acceptance of the task.

An education package was developed in order to create understanding and knowledge of the ageing programme and the method. The target group was broad. The approach was that all personnel related to the facility would carry out the training, which was also carried out prior to the project's handover. The education, for example, discussed up the history of the ageing program, the requirements, basic documents such as SAR and Technical Specifications, and the process of ageing. It included practice examples that are adapted to suit all professions groups, for example, a scoping and screening exercise is done on a car and the ageing management analysis of the car's tires.

For the engineers and staff with practical performance within the ageing program, further training was carried out at a more detailed level, for example in the areas of material knowledge and inspection technology.

2.3. The process of ageing management at CLAB

The ageing process (Fig.1) at CLAB consists of four main parts:

— Scoping and screening;
— Ageing management analysis;
— System analysis (System health review);
— Reporting and follow-up.

The ageing programme also has a strong connection to other activities and documents such as maintenance documentations, maintenance activities, operation and modernization projects.

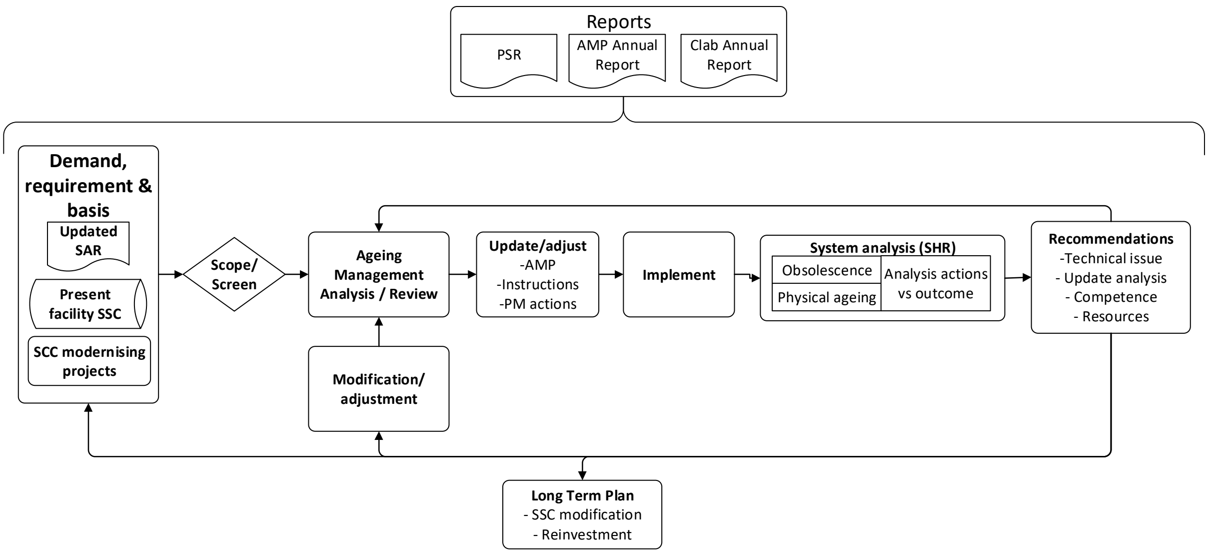

FIG. 1. Process for ageing management programme at CLAB.

The scoping and screening process is a comprehensive work aimed at selecting systems and its components that should be included in the ageing management program. The criterion for a system to be included in the ageing programme is that it is important for radiation safety. This has been defined at CLAB as a system that is of importance for the deep defence, or systems that can cause problems for the deep defence. The work is made more difficult by the fact that classification lists cannot be used. CLAB has equipment without safety significance and some with safety signification with the same security classification. The basic data for scoping and screening is taken from the safety report that relates to the plant's requirements, the plant's description and the plant's design event analysis. Even other requirement documents are taken into consideration for scoping and screening, particularly regarding the systems associated to the physical protection. The screening and scooping is carried out in two stages, at a system level and at a component/functional level.

Systems and components that have been included in the programme then undergo ageing management analysis. The ageing management analysis shows which measures are to be taken to control, minimize and manage the ageing impact, so that actions can be taken to ensure that age-related impairments cannot have any impact on radiation safety. The process follows nine attributes written in NS-G-2.12 [3]: (Scope, Preventive actions, Detection, Monitoring, Mitigating, Acceptance criteria, Corrective actions, Operating experience, Quality management). To get a similarity in the management of generic components, component AMP´s have been written to describe suitable measures, e.g. for electric motors, power electronics and solenoid valves. These measures are used in ageing management analyses for each system. The ageing management analysis concludes with a gap analysis in which the proposed measures are compared with existing measures. When a gap arises, a unit is appointed responsible to correct the gap within a certain time. As the gap is corrected, the analysis is updated, the analysis is completed when no gaps remain.

The system analysis determines the status with regard to ageing (technologically and physically). The input data for the analysis are experiences from operation, maintenance and radiation protection activities and the outcome of the measures indicated by the ageing management analysis. A check is made to see if there is a need to update the system's ageing management analysis. Input for this is new experience, technology development, or if the measures have not been effective enough. How effective the preventive measures have been assessed through a review of the last five years' fault reports. Are there fault reports that show ageing that has not been detected in measures within the ageing program, the programme is updated with these improvements. An important part of the system analysis is the investigation of the system's technological ageing, e.g. availability of spare parts and competence (obsolescence). System analysis is carried out every five years.

A yearly report describes and follows up the previous year's activities of the ageing program. All recommendations from the system analysis are compiled and prepared for decision. Meetings are held with the first line managers to see what action proposals are accepted and can be addressed. Proposals that could not be decided upon or addressed at the first line's management level are lifted up to the plant's management level for decisions. In decision-making, a time is also set when the proposal is to be completed and a responsible person is appointed. When proposals are unaccepted, the reason must be motivated.

2.4. Results from the introduction of the ageing program

The facility consists of approximately 160 systems and about 96 of these are covered by the ageing program. It is important to point out that only those parts of the systems that are of importance to radiation safety are covered. Examples of these systems are various buildings, underground structures, cooling systems, lifting systems, storage pools, storage cassettes and the spent nuclear fuel. The nuclear fuel is included because the facility credits the fuel enclosure as a barrier.

The result of ageing management analysis of all systems in the programme was that 546 gaps were identified where measures were needed to manage ageing. Mostly this was condition checks that were addressed by maintenance, several gaps were also handled by other professionals' groups, for example chemists and operational staff. The gaps involve different types of measures, for example:

- Control of cleanliness in the facility;
- FME program;
- Camera inspections of pipes and pools;
- Extended controls on electric motors;
- Installations of dehumidifiers in underground structures;
- Inspections of nuclear fuel;
- Control of functions;
- Control of chemical parameters, e.g. process water and diesel fuel.

Several unexpected observations were made in connection with the actions being carried out. For example, incorrectly repaired pipes in safety systems (origin from time of construction), foreign objects in storage pools and on fuel that has the potential to cause corrosion problems in the long term, unexpectedly large amounts of microbes in storage pools and leaking bushing in transformers.

Several modernizing projects have been initiated, since the system analysis stated that the systems are exposed to technological ageing (obsolescence). This means that we could have availability problems in the near future due to lack of supplier support or access to spare parts.

The result of the work in the ageing programme has shown that the plant's status regarding physical ageing is good. Technological ageing is a bigger challenge. An important aspect in this context is that new modern components, primarily within I/C, which become obsolete faster than the older technology.

2.5. Practical examples of measure in the ageing management program

2.5.1. FME

Routines and procedures how to work with FME has been implemented. An FME-coordinator has been employed to assist both staff and project to work properly in practical FME-questions. In the picture you can see an FME-covering the unloading pool. This was done prior to reconstruction of the fuel handling machine.

FIG. 2. FME-covering of the unloading pool.

2.5.2. Internal inspection of pipes

One measure in the programme is the internal inspection of selected pipes. One damaged pipe was found during an inspection. The damage was found in a safety system, in a pipe that can supply the storage pools with make-up feed water in case of loss of cooling system. The damage was in an earthquake proof concrete structure which made it problematic to expose the damage for repair. An enlarged control programme has been established for camera inspection of the damage, within the ageing program. No sign of ageing impact has been observed. As long as the damage is stable, it will not be repaired.

FIG. 3. Damaged make up feed water pipe.

2.5.3. Inspections of fuel

Fuel inspections are made with underwater cameras. Fuel elements of different types and ages are selected for inspection and recording periodically with the intention to find changes depending on ageing.

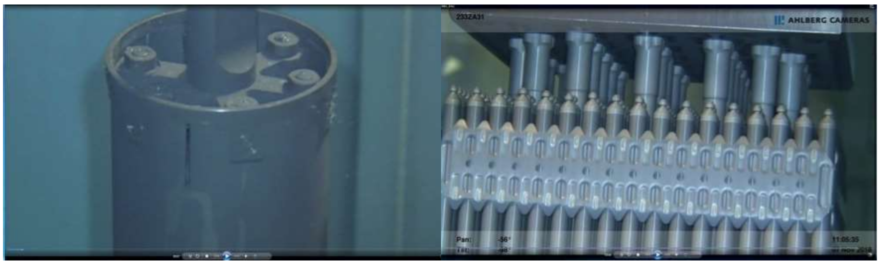

FIG. 4. Fuel inspection pictures.

2.5.4. Measurement of capacitance in capacitors

Capacitor is one of the components that ages in electronic devices, like power supplies and rectifiers. The measure programme has been expanded after implementing of the ageing management program.

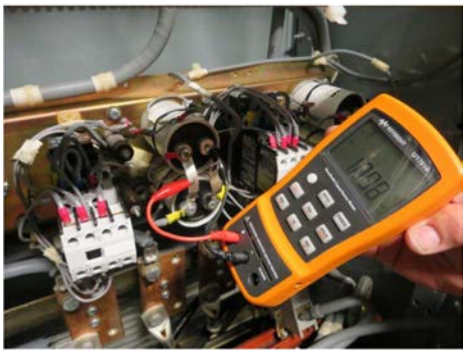

FIG. 5. Measurement of capacitor

2.5.5. Measure of concrete structure

To determine the ageing influence of the concrete structure of the cooling water intake a comprehensive measure programme was performed. There were some thoughts that it could be corrosion in the reinforcing bars. Core drilling was done to analyze the depth of the carbonation process, compressive strength, chloride content and other facts with importance for ageing. In some places reinforcing bars were uncovered to make visual inspections. The measurements showed that the ageing influence of the structure was less than expected, the concrete was in good condition.

FIG. 6. Concrete structure testing.

2.6. Lessons learned

2.6.1. The importance of having clear communication with the supervisory authority

One of the main experiences in connection to the injunction of the ageing programme is the importance of clear communication between the supervisory authority and SKB.

In connection with the remediation of the injunction, a lack of communication resulted in that the authority and SKB had different objectives regarding the extent of what was to be remedied. The cause of the misunderstanding was a poor word selection in SKB's action report. SKB interpreted that a working method was implemented and that the work of analyzing the plant's system had begun with a plan for progress. The supervisory authority intended that analysis of all of the plant's systems should be completed. One contributing factor to the misunderstanding was that SKB interpreted an ambiguity and error writing in the authority's letter to our own advantage.

The differences in the goals were not discovered until SKB sent in the final report of the remediation of the injunction and reported measures taken. The authority responded by issuing an additional injunction regarding the implementation of an ageing program.

An HFE investigation was carried out that showed several reasons, but the main reason is considered to be unclear communication from SKB to authority. Several measures were taken to strengthen SKB's way of working with injunctions, for example clearer role of responsibility between the security department and the line organization and better support for the line organization when handling injunctions.

The experience is to have clear communication with the authority and to clarify any uncertainty.

2.6.2. Level of analysis

When the ageing programme was developed and the first analyses were carried out, considerable time was spent in discussing about the level of work. It is considered important to establish a reasonable level of analysis and measures. If the programme becomes too deep and detailed, it will cost large sums of money and resources. If the programme becomes too general, important aspects of ageing will not be found.

2.6.3. Implementation with in-house staff

The main reason for the success of the work on the ageing management programme is that it was decided that the programme would be prepared by in-house staff. A number of specialists started to work together with the person who should manage the programme in the line organization. As the analysis started, the organization became more involved in the project. This meant that the handover from project to line organization became undramatic. Many of the involved persons didn't notice that work went over to the ordinary organization. Since maintenance and operation personnel had been involved in the analysis from the beginning, no resistance was encountered to perform the measures to reality. Another advantage by doing the work with in-house staff is that the knowledge about the ageing programme is gradually being built up within the organization.

In summary, to do the work with internal resources has given understanding and acceptance to the ageing management program.

2.6.4. Information to decision-makers

System analyses reports are solid information from employees to management in issues related to the plant's status. Problems and malfunctions that were previously discussed in coffee rooms and corridors are now formalized in the system analysis and result in a recommendation on action. Decision on action of the recommendation is then taken by decision makers at different levels depending on the competency of the recommendation.

2.6.5. Documentation

The ageing programme has resulted in a large amount of documentation. For examples, scoping and screening reports, ageing management analysis reports, system analysis reports and assessments on the impact on the ageing programme in connection with modernizing projects. A good document structure and functional tools are required to have good control of the documents and their dependencies. In our case, a person was responsible for the document management. We found that our document management system did not provide required support for the tasks, we were forced to create independent lists and cross-reference tables to be able to manage all the documents.

2.6.6. Development work in the ageing program

A task on how to deal with spare parts in the ageing programme has just begun. The purpose is to have control of spare parts so that they can be installed in the plant without delay. Ageing of spare parts should be postponed or prevented by correct storing and proper maintenance.

After the introduction of the ageing programme it was noticed that manage of equipment with importance for radiation safety is good. Equipment that is important for operational availability is not currently handled within the ageing program. There is a difference in the quality of how these different systems and components are handled. In the future, the aim is to manage equipment important for the operational availability in the ageing management program. A first step to introduce a routine for classification of systems based on priority class has been developed. This considers how important the equipment is for both radiation safety and operational availability. The classification is supposed to replace today's screening for the ageing management program. This enables components of importance for both radiation safety and operational availability to be managed equally. In addition, an opportunity is given to differentiate the significance of a component in different stages. This makes it possible to further increase the focus on important components, while the less important ones can be handled more lightly.

The ageing programme is based on the IAEA Safety standard NS-G-2.12 [3]. The Safety standard was replaced in 2018 by IAEA Safety standard SSG-48 [4]. In 2019, a gap analysis will be carried out to see development needs in the ageing programme due to the updated IAEA document.

2.6.7. Authority inspection of the new ageing program

In 2017, the authorities carried out a renewed inspection of the ageing program. The authority stated that all systems of significance for radiation safety were covered by a functional ageing management program. The authority also stated that there was a commitment among the employees for the work on the program. Some areas of improvement were noted, but the assessment was that the facility met the regulatory requirements very well regarding ageing programs.

3. CONCLUSION

After some failed attempts and some years of hard work we have a working ageing management program. The work has just started, the programme will be managed and improved until the day when the facility is closed. Even though the facility has experienced employs that have worked many years with improvement of maintenance and operational procedures to keep the facility in god condition, the work with the ageing management programme had given us a lot more knowledge about the ageing status and how to deal with it. As a result of the program, the communication and cooperation between different units and levels has increased which has led to an increased efficiency. The programme also gives support to long term planning of both maintenance activities and modifications.

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Paper ID#182
LESSONS LEARNED FROM FUKUSHIMA DAIICHI NUCLEAR ACCIDENT FOR SPENT FUEL STORAGE

K. SHUJI
Tokyo Electric Power Company Holdings, Inc.
Tokyo, Japan
Email: kaminishi.shuuji@tepco.co.jp

T. RITSURO
Tokyo Electric Power Company Holdings, Inc.
Tokyo, Japan

F. TAKATOSHI
Tokyo Electric Power Company Holdings, Inc.
Tokyo, Japan

Abstract

On March 11, 2011, a tremendous earthquake of a 9.0 magnitude occurred in Japan. In the Fukushima Daiichi Nuclear Power Station, fuel assemblies were stored for the Units 1 to 6 Spent Fuel Pool, common pool and dry casks. The paper reports the lessons learned from Fukushima Daiichi Nuclear Accident for spent fuel storage.

1. INTRODUCTION

At 14:46 on March 11, 2011, a tremendous earthquake of a 9.0 magnitude occurred undersea off the coast of the Sanriku region of Japan, triggering a massive tsunami on an unprecedented scale that hit the north-eastern coast 50 minutes later. The earthquake caused the loss of all off-site power supplies of the Fukushima Daiichi Nuclear Power Station (Fukushima Daiichi NPS), but all Units succeeded in cooling the reactors by using emergency power. Units 1 to 3, which were in operation when the earthquake struck, shut down safely as designed.

However, this emergency power was also lost due to flooding from the tsunami, causing the cooling equipment to become inoperable, thereby resulting in the water in the reactor pressure vessels of Units 1 to 3 evaporating into steam. It is supposed that hydrogen, produced by the chemical reaction between fuel rods sticking out of the water and steam, accumulated in the upper part of the reactor buildings and triggered explosions in Units 1 and 3. For Unit 4, it is supposed that hydrogen that flowed in through the joint part of the exhaust stack accumulated when the air in the primary containment vessel of Unit 3 was vented to the outside, leading to the explosion.

Units 5 and 6 were undergoing outage when the earthquake occurred. Cooling via seawater system pump was flooded by tsunami, making it unusable. However, Unit 6 Emergency Diesel Generator (EDG) which was air-cooled and the electricity station distribution system power cable (tie-lines) between Units 5 and 6 had been ensured, temporary seawater system pump etc. were restored and cooling function was ensured. Cold shutdown was achieved for Units 5 and 6 while event progression was controlled.

At the accident, spent fuel assemblies were stored for the Units 1 to 6 Spent Fuel Pool (SFP), common pool and dry casks (see Table 1 and Fig. 1). The paper shows the impacts of the accident and progress status after the accident for spent fuel storage on the Fukushima Daiichi NPS [1].

TABLE 1. NUMBER OF STORED FUEL ASSEMBLIES (copyright: TEPCO, [2])

Storage location	As of March 11, 2011		As of January 31, 2019	
	Spent fuel	Fresh fuel	Spent fuel	Fresh fuel
Unit 1	292	100	292	100
Unit 2	587	28	587	28
Unit 3	514	52	514	52
Unit 4	1331	204	0	0
Unit 5	946	48	1374	168
Unit 6	876	64	1456	428
Common Pool	6375	0	6081	24
Dry Cask (Dry Storage Cask Building)	408	0	0	0
Dry Cask (Temporary Cask Custody Area)	-	-	2,033	0

Note: The number of Units 1 to 3 exclude core loading fuels. The number of Units 5 and 6 include core loading fuels and Unit 6 include transferred fresh fuels from Unit 4 as of January 31, 2019.

FIG. 1. Fukushima NPS site layout (copyright: TEPCO, [2]).

2. THE IMPACTS OF THE ACCIDENT

The tsunami resulted in a total loss of AC power to Units 1 to 5 and common pool, which in turn caused the SFP to lose cooling and supplementary feed function. Furthermore, whereas the D/G for Unit 6 maintained function, seawater pump function was lost so SFP cooling function was lost. The Dry Storage Cask Building also experienced Station Black Out, but the dry storage casks are designed to be air cooled through natural convection.

2.1. Spent fuel pools

Restoring cooling water injection and cooling of the SFP for Units 1 to 6 and the common pool was necessary. In particular, the amount of heat being generated by the SFP for Unit 4 in which all fuel was being stored since the Unit had undergone outage was huge. Because of hydrogen explosions occurred in the reactor buildings of Units 1, 3 and 4, therefore, factors such as access and the ensuing environment made it extremely difficult to achieve cooling water injection and cooling of the SFPs.

Cooling water injection of Unit 4 using concrete pump trucks that extending the boom to inject coolant from right next to the reactor building, began on March 22 and similar operations began at Unit 3 (March 27) and Unit 1 (March 31). Furthermore, since the Unit 2 reactor building did not experience an explosion, so an injection measure that consisted of using a fire engine to inject coolant via pipes inside the building was examined and put into implementation on March 20. Radionuclide analysis of the SFP water provided no data that indicates fuel damage.

As explained above, cold shutdown of reactor Unit 5 and Unit 6 was successful, as was pool cooling on March 19.

2.2. Common pool

Several thousand spent fuel assemblies were stored in the common pool for which it was necessary to restore cooling. The amount of heat generated by each individual spent fuel assembly in the common pool is small, but the vast quantity required a large amount of cooling water injection, therefore, restoration of the cooling equipment installed in the common pool building was required. Off-site power was supplied to the site, enabling the restoration of common pool cooling on March 24.

2.3. Dry casks

At the accident, 9 dry casks had been stored in Dry Storage Cask Building (Fig. 2 and Fig. 3). The tsunami flooded the Building with a large amount of sea water, sand, and rubbles, and the louvers and doors were damaged, but the airflow required for natural air cooling was not inhibited, and there were no cooling problems. The outer appearance of the casks showed no signs of abnormalities concerning soundness. There was no abnormality with result of measuring the dose rate and the temperature as well.

All casks were transported to the common pool, inspected and then necessary parts were replaced. Additionally, the cask which is maximum amount of heat generation was opened for inspection and 3 fuel assemblies were unloaded for external appearance observation at common pool.

The cask was inspected the leakage rate of the primary and secondary lids, the pressure between lids, the monitoring Krypton gas, and the observation of flange surface, metal gaskets for primary and secondary lids, and basket. It was confirmed that there was no problem to the sub-criticality function, the containment function and spent fuel integrity (Fig. 4). Then all 9 casks were stored in the newly constructed Temporary Cask Custody Area.

FIG. 2. Dry Storage Cask Building; photographed August 24, 2011 (copyright: TEPCO, [3]).

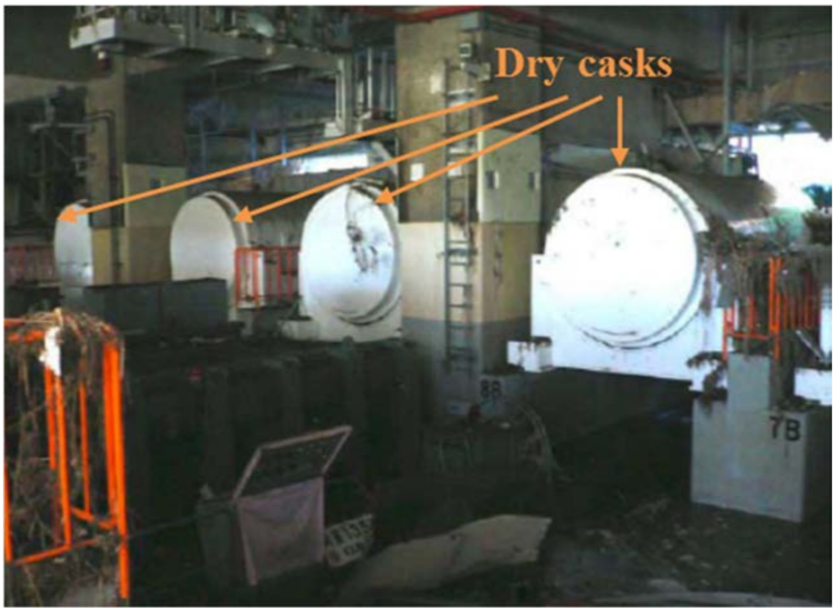

FIG. 3 Dry Casks; photographed March 17, 2011 (copyright: TEPCO, [3]).

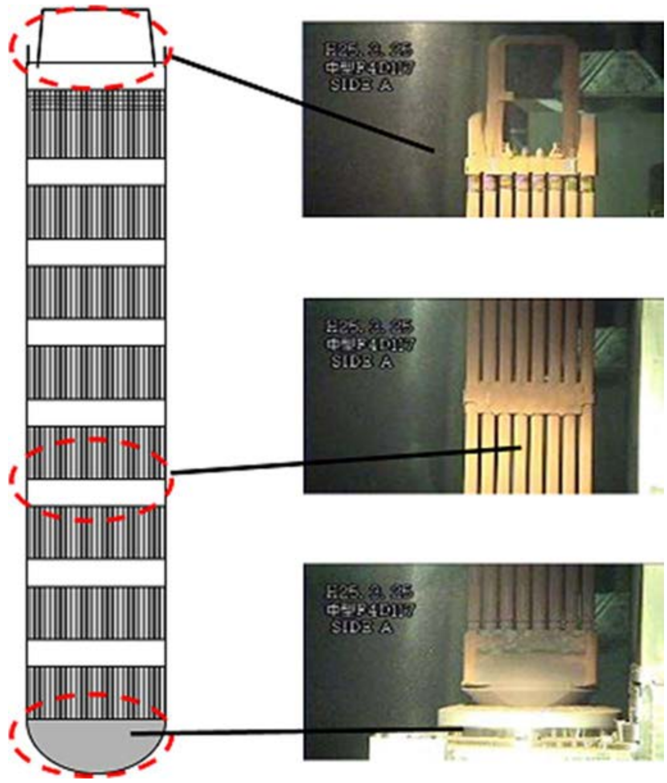

FIG. 4. Result of external appearance observation of spent fuel assemblies (copyright: TEPCO, [4]).

3. PROGRESS STATUS AFTER THE ACCIDENT

About 6000 fuel assemblies, that are stored in Units 1 to 6, are to be removed to the common pool and stored there. However, over 6000 fuel assemblies had already been stored in the common pool prior to the accident. Therefore, the Temporary Cask Custody Area was newly built to store fuel assemblies that were in the common pool in order to ensure capacity (Fig. 5). This facility monitors area radiation, pressure between dry cask lids and temperature of dry cask. Current storable capacity of casks is 50 units and prospective expansion of this capacity is under consideration. The issues of dry cask storage are that it needs to be secured in an area and to be satisfied with the criteria of radiation dose taking effect of the sky shine include stored the contaminated water and the rubbles at site boundary into consideration.

Loading of spent fuel stored in the common pool to dry casks commenced since June 2013. Unit 4 fuel assemblies were removed to the common pool starting on November 18, 2013 and the removal work was completed on December 22, 2014. Work continues toward fuel removal from Units 1 to 3.

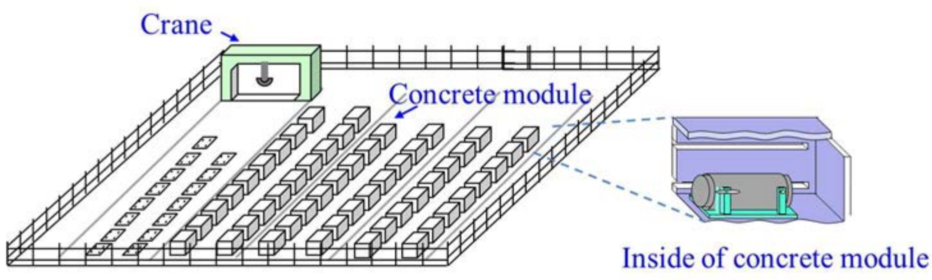

FIG. 5. Temporary Cask Custody Area (copyright: TEPCO, [4]).

4. CONCLUSIONS

Although Units 1 to 3 SFP were affected by tsunami and hydrogen explosion, the stored fuel assemblies were not found with fuel damage. And the dry casks were submerged by tsunami, as the result of inspections, no abnormality was found with safety functions and spent fuel integrity.

Fuel removal from Unit 4 SFP was completed in December 2014. TEPCO will proceed with fuel removal from Units 1 to 3 SFP using common pool and dry casks and with decommissioning work steadily in a stable manner.

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Paper ID#190
OVERVIEW OF REGULATORY FRAMEWORK ON DRY CASK STORAGE IN JAPAN

T. NISHIZAKI
Nuclear Regulation Authority
Tokyo, Japan
Email: takanori_nishizaki@nsr.go.jp

T. MORIMITSU, K. AOKI, H. YAMAGATA
Nuclear Regulation Authority
Tokyo, Japan

Abstract

On March 2019, the regulatory body of Japan, NRA, issued new regulations on dual purpose cask (DPC) for dry storage of spent fuel on the site, along with the licensing process of design certification on DPC. The requirements ensure consistency with those for the interim storage facility off the site and with transport regulations.

1. BACKGROUND

In Japan there are three types of regulation on dry cask with different purposes; for storage on the site, for transport outside the site and for interim storage at a facility off the site where a power reactor is not co-located. These regulations are well-designed separately to fit with the respective situation and seem to be still working in an effective manner, while it is not necessarily taken into account to keep enough consistency between them. The lack of consistency could result in undue complexity and inconvenience in design and operation, even in case to use the applicable cask on different occasions.

In light of the lessons learned from the Fukushima-Daiichi nuclear accident where the dry cask has suffered no significant damage from the unprecedented external event, operators are urged to explore better ways of enabling longer-term storage more safely than in a conventional manner, especially by means of DPC.

2. NEW REGULATIONS

2.1. Dual purpose cask

Since the current regulation on dry cask for use on the site is focused on that for storage, new rules are put in place that regulate DPC capable of adapting to storage and transport arrangements. This would lead to avoid an unnecessary risk at repacking once the spent fuel has been stored into DPC at the site prior to transport. While it remains a matter of choice for operators to decide whether to use DPC for dry storage on the site, the regulatory body encourages operators proactively, but in a non-mandatory manner, to prefer DPC for the prolonged storage on the site with due consideration of the unforeseeable situations.

The regulation on DPC also contains the technical requirements (e.g., confinement, shielding, heat removal and criticality prevention) which are almost the same as those for storage on the site and for interim storage off the site.

2.2. Design-basis hazard of external events

DPC is expected usable for storage at any place and for transport via any transit route. A safety assessment for external hazard, however, should be performed based on site characterization in general. To cope with both, a set of 'reference' external hazard is defined as a design-basis requirement for DPC with the aim to exclude site dependencies. The reference hazard is set up for the key hazardous natural events such as earthquake, tsunami and tornado, and has margin of safety enough to encompass all the external natural hazards postulated for the existing sites. The hazard assessments for other external events are no longer required in case that DPC can meet the reference.

2.3. Design-basis hazard of external events

In order to promote DPC, the regulatory regime is amended for DPC to be granted the design certification. Our licensing process of the design certification consists of both a 'type certification' and a 'type designation' and is applicable for the limited type of target equipment listed in the regulation. The former process is open for whoever is interested, but the latter for domestic and foreign manufacturers, domestic users and importers. The DPC manufacturers can apply for both approvals, independent from users.

Since the storage and transport capabilities come together to be confirmed on a single application, applicants for design certification on DPC are not required to submit any other application for approval.

3. THE WAY FORWARD

Although the technical requirements for DPC are stipulated enough to secure consistency with those for interim storage off the site and for transport, separate applications are required to be submitted in accordance with the relating rules. For instance, in case of diverse use of DPC for interim storage off the site, applicant needs to submit its application to the regulatory body for review and approval which addresses the safety characteristics of the facility based on the site conditions even if the design certification of DPC has been already granted. There still remains room for further endeavour, in particular, to streamline our review processes or administrative procedures.

Paper ID#147
NUCLEAR SPENT FUEL STORAGE: CONCEPTS AND SAFETY ISSUES

F. LEDROIT
IRSN
Fontenay-aux-Roses, France
Email: frederic.ledroit@irsn.fr

A.C. JOUVE
IRSN
Fontenay-aux-Roses, France

I. LE BARS
IRSN
Fontenay-aux-Roses, France

Abstract

The IRSN (Institut de radioprotection et de sûreté nucléaire), the French technical support organization, was asked by the parliamentary inquiry Committee on the safety and security of nuclear installations to provide a report on the concepts and safety issues regarding storage of spent fuel from nuclear power reactors. Based on its expertise in France and on its knowledge acquired during services performed abroad, IRSN examined the concepts of wet and dry spent fuel storage existing worldwide and in France, as well as the associated safety issues. In conclusion, IRSN emphasizes that the choice of a type of spent fuel storage must be assessed with regard to the following considerations. The two types of spent fuel storage, wet or dry, do not completely serve the same needs, as wet storage is essential for spent fuel with high residual heat and dry storage is well suited to highly cooled fuels. In any case, these two types of storage are complementary, but the choice of one or the other largely depends on national choices in terms of spent fuel management (reprocessing or not).

The type of spent fuel (UOX, MOX…) affects the choice of the type of storage, at least for a certain period of time. Thus, spent MOX fuels have a higher residual heat and this decreases less rapidly. Their cooling time before being placed in dry storage is thus much longer than for spent UOX fuels. From the safety point of view, whatever the type of storage, the key parameter is the residual heat of the spent fuel to be stored. In this respect, wet storage, which is generally used for spent fuel with higher residual heat, requires more extensive safety provisions than dry storage where safety relies on passive systems. IRSN also considers that a particularly important point for the safety of spent fuel management operations is the control of zirconium fuel cladding ageing, which depends on the storage temperature. On this point, wet storage offers guarantees whereas, in dry storage, the ability to directly and easily examine fuel cladding is reduced.

1. BACKGROUND

In December 2017, the French National Assembly created a Committee of inquiry into the Safety and Security of Nuclear Facilities. The creation of the Committee was decided after several intrusions of non-governmental organizations into the premises of nuclear power plants, the last targeting spent fuel storage pool security.

During the spring of 2018, the Committee, composed of about 30 parliamentarians, alternated hearings in the Assembly offices and visits of nuclear sites. The Committee's initial work had revealed that spent fuel management presents particular issues. The operation of nuclear power reactors leads to the generation of spent fuel, which then has to be stored for a period of time dictated by national choices regarding the management of radioactive materials and waste (reprocessing/recycling, long term storage, etc.). In this context, the Committee learned of EDF's plan to build a centralized spent fuel pool facility, designed to store spent fuel for a period of around one hundred years. It also found that storage in a pool is not the only option and that an increasing share of spent fuel in many countries is put into 'dry' storage using large containers (or 'casks').

Therefore, on 26 March 2018, the Chair of this Parliamentary Committee of Inquiry wrote to the French Institute for Radiological Protection and Nuclear Safety (IRSN – Institut de radioprotection et de sûreté nucléaire), the French technical support organization, to seek its technical opinion on the nuclear safety issues associated with a strategy for managing irradiated nuclear fuel (also known as spent fuel) based on the storage of that fuel only in a pool (or underwater so called wet storage) or also in dry storage facilities.

Based on its expertise in France and on its knowledge acquired during services performed abroad, IRSN examined the concepts of spent fuel storage existing worldwide and in France, as well as the associated safety issues, taking into account the characteristics of different types of fuel and the various types of storage (wet or dry, on-site or centralized).

The objective of this presentation is to introduce the main findings of the report submitted by IRSN to the Committee in June 2018. The IRSN report is published in English on IRSN Website [1].

2. INTRODUCTION TO STORAGE

It should be first reminded that storage is a facility dedicated to the temporary holding of radioactive material or waste. This supposes that the retrieval and the transport of the material or waste could be done after the period of storage. Therefore, sufficient provisions should be taken regarding traceability and transport means. Whatever the storage concept is, four fundamental safety functions have to be ensured, both for normal conditions and accidental situations: protection against radiation exposure, sub-criticality of fissile material, confinement of radioactive material and cooling of radioactive material.

3. FUTURE OF SPENT FUEL ASSEMBLIES

Spent fuel from nuclear power plants requires interim storage after being unloaded from the reactor. Its initial residual heat is too high. So, decay of the radioactivity that it contains, which gradually reduces this heat, is necessary to enable it to be transported and managed using the chosen method. In all cases, it is stored initially in the reactor spent fuel pool. Then, depending on the chosen management option (reprocessing or disposal), two practices are used throughout the world.

If the spent fuel is to be reprocessed (as it is in France, Japan and Russia), the reprocessing plants have pools to store it before reprocessing (generally during five to ten years after it is unloaded from the reactor). The use of this type of storage is essentially linked to the processes of these plants, the pools in which the fuel is placed being directly connected to the reprocessing workshops. In addition, the capacity of these pools is generally very large to provide a buffer between activity at the reactors and activity at the plant and to allow additional cooling. Once they are separated, uranium and plutonium are sent for recycling into fuel assemblies made from plutonium (MOX) or from enriched reprocessed uranium (ERU). The storage methods for spent MOX and ERU fuels then depend on the planned future of these fuels in the countries concerned.

If spent fuel is not reprocessed (as in most places in the world), the unloaded fuel is generally placed in dry storage facilities once it has cooled sufficiently in a pool. Current storage concepts are based on the average residual heat of fuel assemblies being around 2 kW. To a certain extent, it should be possible to adapt these concepts.

As illustrated on Fig. 1, the residual heat per unit of the fuel assemblies to be stored is a decisive factor in determining the type of storage to be used.

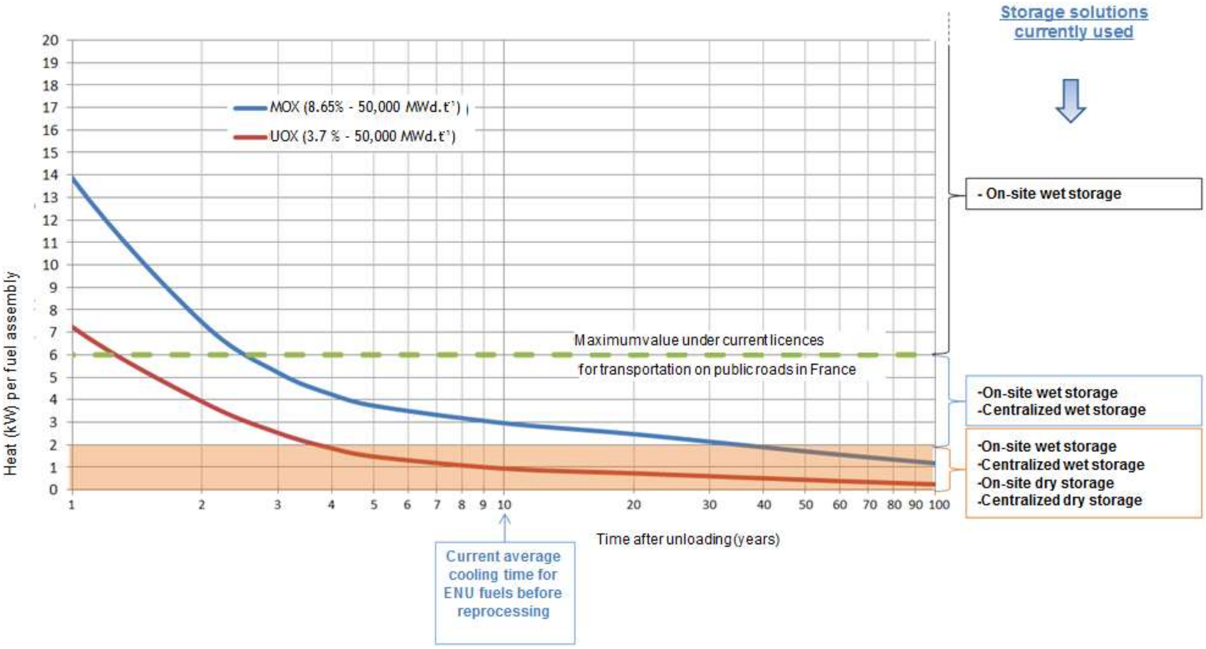

FIG. 1. Suitability of storage solutions based on the residual heat of the spent fuel.

So, storage in a pool is essential for spent fuel that has been recently unloaded and dry storage is suitable for fuel that has cooled significantly.

In any case, the two types of storage are complementary, but the decision to use one or the other after an initial cooling phase, of necessity in a pool, depends to a large extent on national choices regarding spent fuel management.

4. SPENT FUEL MANAGEMENT IN FRANCE

In France, the decision to store spent fuel in a pool is linked primarily to the choice to reprocess spent fuel for recycling plutonium (24 of the 900 MW reactors are currently licensed to use MOX fuel) and uranium (four 900 MW reactors are currently licensed to use ERU fuels).

After being unloaded from the reactor, ENU spent fuels are:

— Stored in a reactor pool until their characteristics are compatible with transportation to the Orano Cycle site at La Hague, particularly when their residual heat is around 6 kw per fuel assembly with current casks and under current transport licences;
— Stored in pools on the Orano Cycle site at La Hague until they are reprocessed, which happens approximately 10 years after the end of their irradiation in a reactor.

Spent ERU and MOX fuels are managed in a similar way, but their reprocessing is differed. Pending a decision about their future, EDF plans to create a centralized storage pool to store spent MOX and ERU fuels.

Spent ERU fuels have similar characteristics to spent ENU fuels. The ENU fuels currently used by EDF could, with the current concepts, be stored in dry conditions after cooling for around five years. However, because of the amount of time remaining before they are reprocessed, there seems to be little point in using this type of storage. If a spent fuel reprocessing plant were to be unavailable for a long period (eventually causing saturation of the existing storage capacity), using this type of storage could be one solution.

Fresh MOX fuels loaded into a reactor have a high plutonium content to give them an equivalent burnup to that of the ENU fuels used with them in the reactor. Due to this plutonium content and its isotopic composition, spent MOX fuels have a higher residual heat. Because of their higher transuranium element content, their residual heat is also slower to decay. The cooling time before they can be placed in dry storage is therefore substantially longer than for spent ENU fuel, i.e. it takes several decades to reach a residual heat per fuel assembly of 2 kW. The use of dry storage could therefore be envisaged only beyond this period of time.

5. ASSETS AND LIMITING FACTORS OF WET STORAGE

It is obvious that wet storage implies the use of pool to store the spent fuel underwater. Taking a closer look at the existing concepts of pools, it could be mentioned that such storage could be located:

- Above-ground as it is the case in nuclear power plants to facilitate loading/unloading of fuel;
- Semi-underground: in that case, the water level is close to the ground level;
- Underground such as in the central interim storage facility for spent nuclear fuel (named CLAB) in Sweden.

Two types of storage modes of spent fuel assemblies could be observed either in racks or in baskets. Wet storage is particularly suitable for fuels with a high residual heat, which can therefore not remain in air without deterioration of their cladding. Water is an effective coolant and active cooling systems that use it can keep fuel cladding at low temperatures. In addition, a pool has considerable thermal inertia, making it easier to deploy emergency systems if the cooling systems are lost.

The main safety requirements for wet storage are to maintain a sufficient water inventory in the pool and to have cooling systems available in all plausible circumstances. Because of the high residual heat of the spent fuels contained in the pool, a prolonged loss of cooling without water makeup could have very significant consequences for the environment, with it becoming impossible to go near the pool because of the high dose rate induced by the fuel in the absence of any attenuation of the radiation by water.

Consequently, a spent fuel pool, particularly if it receives spent fuel that has hardly cooled, must be of a particularly robust design — with sufficient margins to cope with any risks that can be envisaged — and its operation must allow appropriate monitoring of both the installation itself and the fuel it contains.

Experience feedback from the Fukushima accident lead safety approaches for controlling these risks to be reinforced, aiming to maintain a sufficient water inventory in extreme situations of natural origin.

Current industrial techniques enable pools to be built that control the risks of fuel uncovering, with the buildings housing the pool providing protection against external hazards (particularly the aircraft shell).

It generally takes about a decade to build a facility of this kind, based on current experience feedback from nuclear facilities built in France.

6. ASSETS AND LIMITING FACTORS OF DRY STORAGE

There is a wide variety of dry storage types existing worldwide. As for the wet storage facilities, dry storage could be above-ground, semi-underground or underground. These various types could be summarized into three main categories:

- Storage in wells in concrete structures with a plug;
- Storage in casks comparable to transport packages but used for dual purpose (transport and storage);
- Storage in silos which could be either horizontal or vertical; for this concept spent fuel is placed into a canister that is introduced in a concrete structure.

Dry storage is reserved for fuel that has cooled sufficiently (to around 2 kW on average per fuel assembly with current concepts). Consequently, it has the advantage of generally using passive cooling systems, which limits operating constraints, and it lends itself particularly well to modular construction, adapting to needs or even enabling old modules to be replaced over time.

The safety requirements are the maintenance of passive cooling and the quality of the containment barriers between the radioactive materials and the environment.

This type of storage has the advantage of a simpler, more robust design and less operational intervention. Depending on the design, direct monitoring of the condition of the fuel cladding (the first containment barrier), which is subject to the most demanding thermal conditions, is generally not possible.

In any case, if an accident happens, the smaller number of fuel assemblies and their lower residual heat will mean fewer consequences for the environment.

It generally takes around five years to build this type of facility, depending on its modularity and whether or not existing cask concepts and support installations are used.

7. STORAGE FACILITY LIFETIME

Moreover, regardless of the type of storage, significantly longer storage periods than the usual periods (of a few decades) will require the definition of appropriate requirements (particularly in terms of structure design and of safety margins).

8. CONTROLLING THE AGEING OF ZIRCONIUM FUEL CLADDING

For IRSN, one particularly important point for the safety of spent fuel management is controlling the ageing of zirconium fuel cladding, which depends on storage temperature. This cladding is the first containment barrier for the radioactive materials. In addition, its mechanical strength is important for the operations to take place after storage (transport, reprocessing or disposal).

Wet storage offers guarantees in this respect, given the low temperatures and the potential for direct examination of cladding. Countermeasures (canisters for defective fuel) can also be taken if ageing phenomena are detected. There is a significant experience feedback available in France and throughout the world on the behaviour of cladding underwater, at least for periods of a few decades.

With dry storage, it is more difficult to examine fuel cladding directly. Any inspections made are at best indirect (no release of gases into the cask cavity, etc.), or impossible (fuel canisters sealed by welding constituting the second and final confinement barrier); they do not enable the detection of ageing mechanisms.

Any guarantees that the ageing of cladding is controlled are based primarily on studies, which have notably defined the maximum acceptable temperature for cladding in storage. No examinations of fuel carried out to date, as far as IRSN is aware, have challenged the findings of these studies. However, many studies are ongoing. Moreover, there is limited information available for fuels with a high burnup (more than 45 GWd/t), for MOX fuel (especially with a high initial plutonium content) and generally for long storage periods (more than 40 years).

9. CONCLUSION

To conclude, IRSN considers that decisions about the type of storage to be used for spent fuel must be assessed in the light of the following considerations.

The two types of spent fuel storage that could be envisaged (wet or dry) do not serve the same needs, since wet storage is absolutely necessary for fuel that has hardly cooled and dry storage is suitable for fuel that has cooled substantially.

The type of spent fuel (ENU, MOX or ERU) affects any decision about which type of storage to use, at least for a certain period of time, because MOX fuels have a higher residual heat for longer.

From a safety point of view, regardless of the type of storage, the decisive parameter is the residual heat of the fuel to be stored. Wet storage, which generally contains hotter fuel, requires more substantial safety measures than dry storage, for which more passive measures can be implemented. In dry storage, cladding (the first containment barrier) is subject to greater thermal stress and is more difficult to inspect.

REFERENCES

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3.3. TRACK 3 – TRANSPORTATION IN THE BACK END

Overview prepared by T. Tate (United States of America) and J.P. Carreton (France), **Track Leaders**

The track highlighted experience in the USA with planning for transportation following the shutdown of a site. The presenter discussed the importance of preparation activities just before the site closure and immediately after. An important lesson learned is the building of relationships with the people on the site for information. The track also included discussion on the impact of higher burnup fuels on the necessary cooling time in spent fuel pools prior to transferring to dry storage. Various modelling approaches were discussed for comparison of conservative modelling results against realistic modelling results. The experience with the transportation of spent fuel in France was highlighted. Currently, Orano conducts about 200 shipments per year. The need for managing a fleet of railcars and transport trucks was discussed due to the varying situations and conditions encountered that impacts transport. Efforts taken in the USA to support and prepare transportation of high burnup fuels was discussed, including considerations on radionuclide inventory, internal pressures, and cladding performance. Extensive test of high burnup fuel that has undergone hydride reorientation and their impact on the cladding stress was discussed. Challenges with the dry storage and transportation of high burnup and damaged fuels in Spain were also discussed. Currently, transport of spent fuel in Spain is limited to less than 45 GWd/MTU; however, there are efforts underway to remove this limitation. Construction of an interim storage facility in Spain has begun but is currently paused. Preparations for the transportation and disposal of spent nuclear in Finland were discussed. Disposal of fuel in Finland is expected to begin in the 2020's. The interim storage facility is adequate for existing spent fuel in Finland.

This track also highlighted the experience in Russia with an international multimodal transport of spent nuclear fuel, which includes research reactor spent fuel from 13 countries. To support shipments, a special semi-trailer was designed. The packages necessary to support road, rail, air, and sea transport were discussed during the presentations. The United States highlighted a collaborative international multimodal spent nuclear fuel transportation of three surrogate PWR assemblies from Spain to the USA. Data were collected during all modes of transportation. Real transport data collected compared with data from 125 tests showed that the tests are bounding. The experiment concluded that handling activities provided the biggest strain on the fuel. Additionally, the evolution of transport regulations for spent fuel was discussed, including the challenges with implementation of the regulations. One of the most significant changes occurred in 1964 when mechanical test requirements were introduced. Future challenges included demonstrating compliance for fuel designs for longer reactor cycle time and higher burnups and maintaining transportability with the trend towards the use of dual-purpose casks. Finally, experience with the transportation of sensitive nuclear materials and spent nuclear fuel in the UK was discussed. It was noted that there has never been a release of nuclear materials during the performed domestic or international transports. The design features and licensing criteria for the International Nuclear Services (INS) ship vessels used in the transport was presented in the Conference. The UK has a comprehensive approach for the security of transport of nuclear materials in both domestic and international transport.

Session 3.1: Transportation in the back end (part 1)

Session Chairs: M. Golshan (United Kingdom) and B. Ficker (Hungary)

Session 3.1 comprised of six presentations, two from the United States of America and one from Armenia, one from France, one from Spain, and one from Finland.

- **Paper ID#12 by S. Maheras (USA)** presented the experience with evaluations conducted for the removal of spent fuel from 14 shutdown reactor sites to support planning for future removal activities. Importance of understanding the inventory and conditions of the site was highlighted. Included in discussions were lessons learned and several common themes identified such as importance of safety, preparations for site visits, questions submitted to sites, documentation of conditions, and the importance of compiling notes. Additionally, the collection of data, identifying fuel inventory issues, and capturing data shortly after shutdown were discussed. Stakeholder engagement was also discussed.
- **Paper ID#75 by S. Bznuni (Armenia)** presented the challenges associated with higher initial enrichment fuels on the storage and transportation on the back end of the fuel cycle. Impacts such as higher decay heat and neutron/gamma doses on the precooling times needed in spent fuel pools and the burden on loading and transport.
- **Paper ID#108 by Y. Solignac (France)** presented the experience of Orano TN with the successful transportation of spent nuclear fuel. Discussions on establishing a robust programme based upon extensive lessons learned were included. France was highlighted as an example of establishing strong collaborations between shippers and nuclear power plant operators.
- **Paper ID#111 by R. Torres (USA)** presented studies performed to ensure spent fuel cladding performance during the transportation of high burnup nuclear fuel. Issues such as hydride reorientation and required data necessary to support licensing activities were discussed. Age-related mechanisms that have potential to challenge cladding were also discussed.
- **Paper ID#124 by A. Palacio Alonso (Spain)** presented Spanish activities associated with its open cycle strategy for the back end of the fuel cycle including removing licensing challenges with high burnup fuel and challenges associated with loading damaged fuel. Considerations of challenges with current casks in Spain for future transportation of spent fuel to the centralized storage facility.
- **Paper ID#128 by J. Tuunanen (Finland)** presented Finland's preparations for the construction of an encapsulation plant and spent fuel disposal tunnels in preparation for the submittal of an operating license application for a disposal facility.

Session 3.2: Transportation in the back end (part 2)

Session Chairs: J. Heinonen (Finland) and J.P. Carreton (France)

Session 3.2 comprised of 4 presentations, one from Russian Federation, one from the United States of America, one from IAEA, and one from the United Kingdom.

- **Paper ID#177** ***(Invited)*** **by A. Leshchenko (Russian Federation)** presented Russia's experience with the international multimodal transport of spent nuclear fuel from research reactors. The programme has completed shipments from 15 countries to Russia. The presentation highlighted the key engineering and logistical solutions required to successfully organize and complete the shipments. Issues involving the need to improve harmonization of requirements were identified.

- **Paper ID#184 *(Invited)* by S. Saltzstein (USA)** presented the experience of the United States of America with an international multimodal spent nuclear fuel transportation test project. The project involved the shipment of 3 PWR surrogate fuel assemblies from Spain to the USA. The presentation highlighted the international collaboration involved with ensuring this successful project. The presentation also highlighted the data collection during the project and presented information regarding the comparison of the collected transport data and testing data. It was noted that handling activities places the highest stress on the fuel.
- **Paper ID#205 *(Invited)* by S. Whittingham (IAEA)** presented the history and evolution of transport regulations for spent fuel. The presentation highlighted key changes that have occurred in the regulations that have improved the safety of spent fuel transportation. The presentation also highlighted challenges in fuel designs and transportation package designs that will impact storage and transportation of spent fuel in the future.
- **Paper ID#196 *(Invited)* by M. Crowther (United Kingdom)** presented the experience of the International Nuclear Services (INS) with the transportation of sensitive nuclear materials, spent fuel, and other nuclear materials. The presentation included discussions on the key design features of the shipping vessels and the licensing of the transport packaging. The extensive security programme was also highlighted as an important aspect associated with the success of the program.

Paper ID#177
INTERNATIONAL MULTIMODAL TRANSPORT OF SPENT NUCLEAR FUEL THROUGH THE EXAMPLE OF RESEARCH REACTOR SPENT FUEL RETURN PROGRAMS

S.V. KOMAROV
Sosny R&D Company
Dimitrovgrad, Russian Federation
E-mail: office@sosny.ru

A.A. IVASHCHENKO
Sosny R&D Company
Dimitrovgrad, Russian Federation

Abstract

The paper describes key engineering and logistical solutions for organization of international shipments of nuclear materials from research reactors. The evolution of transport equipment and routes is generalized, and the issues requiring harmonization of national requirements and procedures for the safe transport of the spent nuclear fuel are identified.

1. INTRODUCTION

The International Programme on Russian Research Reactor Fuel Return (RRRFR) is coming to its completion giving an opportunity to summarize the lessons learned.

The Programme has completed nuclear material shipments from 15 countries to Russia; this required solving a wide range of various tasks associated with preparation and transportation of nuclear fuels of different health with poor infrastructure available involving all transport modes to minimize transiting through third countries. The unique engineering solutions developed for these conditions are of interest to future nuclear fuel shipment projects.

The experience gained by Russian companies from the RRRFR Programme is demanded in the current international MNSR HEU Take-Back Program. In 2017 and 2018, international SNF shipments from Ghana and Nigeria by air took place.

2. TECHNICAL SOLUTIONS FOR RR SNF SHIPMENTS

The choice of equipment and techniques for preparation of the spent nuclear fuel of any type of reactors for transportation is dictated, firstly, by the fuel type and health and, secondly, by the method of its storage. Preparation of the spent fuel for transportation is aimed at making it loadable and transportable to the consignee's site. As a rule, this is done through repackaging, i.e. placing the spent fuel in special canisters, which in their turn are loaded into baskets and, then, into casks.

In 2006–2010, a large IAEA project on Russian-origin research reactor SNF removal from Vinca Institute, Serbia was completed. The big quantity of fast-degrading spent fuel required that the fuel be removed as a single run and in the shortest possible time. What made that campaign the most complicated RRRFR project is the transit through Hungary and Slovenia, a long route over the Mediterranean Sea and the Atlantic Ocean, a variety of transport modes (road, rail and sea), two types of shipping casks (TUK-19 and SKODA VPVR/M), new European regulations, and a lot of other nuances [1].

In preparing for the SNF removal, Sosny engineers developed and justified safety of the damaged SNF removal from the packaging, in which they had been stored for a long period of time, as well as repackaging for interim storage and transport. A set of tools and accessories (more than 150 types) — from relatively simple long-length grapples of different design to sophisticated equipment for opening the primary SNF package — was developed and fabricated for handling the SFAs and canisters. In addition, a new large-capacity packaging was developed and fabricated for the transport of the spent fuel in the TUK-19 and SKODA VPVR/M casks (Fig. 1).

Sosny engineers and other Russian experts analyzed and justified all safety aspects of handling the new canisters. To prevent explosive hydrogen and oxygen concentrations, the choice fell on an 'untight' design for the canisters that allowed a regular blow-down of the spent fuel in the cask [2].

FIG. 1. 'Untight' canisters and equipment for repackaging of the damaged Vinca SNF.

The Serbian removal project became one of the most complicated in terms of the SNF shipment licensing. Analysis of various routes for their feasibility, safety, and cost-effectiveness revealed the most acceptable one, i.e. transit through Hungary and Slovenia to a Slovenian port in the Adriatic Sea, a sea section to the Russian port of Murmansk, and then a rail shipment to the reprocessing plant. The selected route was supported by all necessary authorizations, i.e. Russian certificates for the package design and transportation endorsed in Serbia, Hungary and Slovenia and licenses for transit through Hungary and Slovenia. In December 2010, almost 2.5 tonnes of the Serbian spent fuel were delivered to Mayak PA.

The TUK-19 and SKODA VPVR/M casks used in the RRRFR programme fit well for handling on most of the RR sites. However, some projects required development of additional equipment to enhance safety of the SNF reloading operations. For instance, the Romanian research reactor VVR-S did not allow the existing SNF loading technologies, i.e. underwater or "air" reloading of the spent fuel into TUK-19 casks. So, it was decided to develop a special transfer cask with an automatic grapple for handling the SFA-containing basket [3]. The dry run and subsequent SFA reloading operations proved safety of that technology both under normal and accident conditions. Similar technologies were developed for the follow-up projects involving SKODA VPVR/M casks when the standard "bottom" loading seemed impossible (Fig. 2).

FIG. 2. Loading canisters with liquid SNF into SKODA VPVR/M through the transfer cask.

Another example of development of a specific technical solution was preparation for removal of the liquid irradiated high-enriched nuclear fuel (LSNF) from the INN-3M research reactor, Uzbekistan [4]. In spite of the fact that Russia has reactors consuming the liquid nuclear fuel, no technology for the LSNF transport had ever been developed. In addition, uranyl-sulphate water solution was not on the list of the fuels reprocessed at Mayak PA.

Preparation for the removal included development of special equipment for the discharge of the liquid spent fuel from the reactor into temporary storage canisters and, then, into transport canisters and a SKODA VPVR/M cask (Fig. 3) For the LSNF delivery to the reprocessing plant, a transport plan and a fuel receipt, temporary storage and reprocessing procedure were developed, equipment for the LSNF discharge from the transport canister into the reprocessing line was fabricated, and calculations of design, fire and explosion risk, nuclear and radiation safety during receipt, temporary storage and reprocessing were performed.

FIG. 3. Equipment for preparation of IIN-3M LSNF removal.

3. EVOLUTION OF RRRFR MULTIMODAL SHIPMENTS

The first RRRFR shipment of the spent fuel took place in 2006. The VVR-SM SFAs were transported from Uzbekistan to the Radiochemical Plant in the TUK-19 casks by rail in the TK-5 container railcars, and from the Institute to the rail terminal — by road. The road section of the route required developing special equipment that could not be used in other projects.

Later, financed by the U.S. DOE, 16 SKODA VPVR/M casks were fabricated. The Czech-made SKODA VPVR/M cask fit better for multimodal shipments, since a 20 ft freight ISO container could easily accommodate it, and the ISO container handling procedure is standard for nearly all modes of transport. Using the SKODA VPVR/M cask for the SNF shipments in the Russian Federation required getting a certificate for the package design and shipment and adapting the Mayak PA's container receipt procedure. Successful certification of the package and preparation of the reprocessing plant's infrastructure were followed by a number of shipments of the spent fuel in the SKODA VPVR/M casks by road, rail and water [5].

When implementing the Romanian VVR-S SFA removal project, there emerged a necessity to use air transport due to the problems encountered with land transit through third countries. Russian experts had to apply efforts to, first, justify safety of air shipment, and, then, to arrange it. Since the activity of all 70 S-36 SFAs as of the date of the shipment did not exceed the value of 3000 A2 specified in the NP-053-04 and TS-R-1 regulations, the air shipment of the SFAs in a B(U) type package was possible; so, the choice fell on the TUK-19 cask. A transport overpack based on a large-capacity 20 ft ISO container (Fig. 4) accommodating three TUK-19 casks tied down with turnbuckles and capable of withstanding accelerations and vibrations typical of all transport means [5] was developed to ensure multimodality of the shipment.

The dynamic deformation and strength analysis performed by Russian experts for the TUK-19 cask under impacts simulating normal and accident transport conditions including an aircraft crash, as well as nuclear risk analysis for the worst cases of impact onto a solid target demonstrated that the air shipment of the VVR-S SFAs in the overpack satisfied safety requirements of the NP-053-04 and TS-R-1 regulations. The flight over the Black sea to avoid transit countries minimized air transport risks [6].

Successful air shipments of B(U) type packages by An-124-100 aircraft from Romania and Libya in 2009 demonstrated a possibility in principle to transport the SNF RR by air. The next step forward in this direction was development of a Type C package for air transportation of various radioactive materials of unlimited activity.

FIG. 4. Overpack for multimodal transportation of TUK-19 casks.

In 2009, US DOE/NNSA addressed Sosny R&D Company to analyze a possibility to develop a Type C package based on the SKODA VPVR/M cask. During the pre-conceptual study it was decided to develop an impact absorber to take in a part of the energy during an impact of the cask onto a target. The analysis of various impact absorbers revealed the most promising option, i.e. a two-piece cylinder with a flange joint in the middle and hollow titanium balls as absorbing elements arranged inside. The overpack was assigned identification number TUK-145/C.

The Russian and international regulatory requirements for Type C packages do not impose additional restrictions on the content radioactivity but require maintaining the integrity after an impact at a velocity of not less than 90 m/s and exposure to fire for a period of one hour. Compliance with these requirements was verified by model analysis of possible air crashes and physical tests of a 1:2.5-scale mockup TUK-145/C package.

The Russian certificate of approval for the TUK-145/C package design granted in 2012 allowed air shipments of the RR spent fuel from Vietnam and Hungary (2013), and the liquid fuel from Uzbekistan (2015) [7].

The development of the Type C package made the air transport the main mode for international shipments of the RR spent fuel (Fig. 5). It allows optimizing physical protection and emergency response, avoiding transit issues, shortening the time of the shipment and ensuring the highest safety level required for the packages of radioactive materials by the IAEA safety standards.

FIG. 5. TUK-145/C for air transport of the spent fuel.

4. APPLICATION OF SHIPMENT ORGANIZATION EXPERIENCE IN MNSR CONVERSION PROGRAM

The Chinese-design 27 kW miniature neutron source reactors (MNSR) are research reactors used, mainly, for neutron activation analysis, education and training. The reactor core contains about 1 kg of 90% enriched HEU. In 2006, the IAEA initiated a coordination research project (CRP) to support conversion of such reactors to LEU fuel. At the 2016 Nuclear Security Summit, the USA and PRC committed themselves to joint effort with support from the IAEA to complete conversion of the MNSR facilities in Ghana and Nigeria to the LEU fuel within the shortest possible time. The PRC marked itself ready to convert all other Chinese-origin MNSR facilities to the LEU fuel [8].

In 2017, the GHARR-1 research reactor in Ghana was converted to the LEU fuel, while the HEU fuel was subject to return to China. TUK-145/C fitted with a new basket and certified as TUK-145/C-MNSR was used for the shipment of the core and several non-irradiated fuel pins. An automatic remotely-controlled transfer cask for the core reloading from the reactor into the shipping cask and auxiliary equipment were developed for the safe SNF handling on the reactor site (Fig. 6).

FIG. 6. Preparation for removal of the MNSR SNF from Ghana to China.

In December 2018, the second MNSR campaign, i.e. removal of the spent HEU fuel from Nigeria, took place. Specific NIRR-1 infrastructure, administrative and security requirements for the removal required developing and implementing relevant engineering solutions to modify the technology and equipment for reloading of the fuel from the reactor core into a shipping container, personnel training, and the removal management in Nigeria.

5. HARMONIZATION OF INTERNATIONAL APPROACH TO SHIPMENT ARRANGEMENTS

The following aspects should be noted in analyzing experience in arrangement of international RR SNF shipments:

— In general, the worldwide trend to unify package design and transport conditions requirements on the basis of the IAEA recommendations has a positive effect on development of international shipments from the viewpoint of broadening options and simplifying licensing procedures.
— In addition to the IAEA recommendations, many countries impose national requirements adding complexity to administrative approval procedures, for instance:
  - Russia: ban on NM shipments by passenger aircraft, a mandatory specific regulator's authorization
  - (license) for NM shipment by air transport, approval of the design of the Type A package with non-fissile material;
  - France: an additional regulator's authorization for certain air shipments over the territory of France;
  - Vietnam: one and the same authorization procedure (in particular, the prime minister's written authorization) for any radioactive material shipments including empty packaging (OOH2908);

- China: design approval for some cases, when the IAEA recommendations do not require multilateral approval, for instance, for Type B(U) package containing fissile-excepted radioactive materials (OOH2916).

— The multilateral approvals of package design and shipment certificates imposed by the IAEA regulations were obtained in each country in a variety of ways.

— The unified certificates for package design and shipment issued in Russia are never endorsed in the countries involved. Most of the countries endorse the Russian certificate as "multilateral approval" of the certificate for the package design and issue a separate certificate for the package shipment in compliance with national administrative procedures. The certificate of shipment approval goes under different names and formats in different countries.

— National administrative procedures for getting an OVF permit for a chartered plane carrying a radioactive cargo can differ significantly in different countries. In addition to a regular set of documents (air carrier's documents, a certificate of package design, liability insurance, etc.), additional documents may often be required, for instance: endorsed certificate of the package design in the countries of departure and landing, import and export licenses, OVF permits from all countries of transit, departure and landing.

6. CONCLUSIONS

High Enriched Uranium Take-Back Programs have really boosted the development of the SNF safe handling technologies and the SNF transport equipment.

Experience in multimodal shipment arrangements and the engineering solutions developed by Sosny R&D Company are universal and can be applied (and have already been applied) in any projects involving handling the spent nuclear fuel from the research, power and propulsion reactors.

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INTERNATIONAL MULTI-MODAL SPENT NUCLEAR FUEL TRANSPORTATION TEST: THE TRANSPORTATION TEST TRIATHLON

E.A. KALININA, D.J. AMMERMAN, C A. GREY, M. ARVISO, C. WRIGHT, L. LUJAN, S.J. SALTZSTEIN
Sandia National Laboratories,
Albuquerque, NM, USA
sjsaltz@sandia.gov

S.B. ROSS, N.A. KLYMYSHYN, B.D. HANSON
Pacific Northwest National Laboratories
Richland, WA, USA

A. PALACIO ALONSO, I. FERNANDEZ PEREZ, R. GONZALEZ GARMENDIA, G. CALLEJA CALATUYUD
Equipos Nucleares, S.A., S.M.E. (ENSA)
Maliaño, Spain

W. CHOI
Korea Atomic Energy Research Institute
Daejeon, Korea

Abstract

Can Spent Nuclear Fuel withstand the shocks and vibrations experienced during normal conditions of transport? This question was the motivation for the multi-modal transportation test conducted in June-October 2017. In this project the US Department of Energy (DOE) (through Sandia National Laboratories and Pacific Northwest National Laboratory) collaborated with the Equipos Nucleares SA, SME (ENSA), Empresa Nacional de Residuos Radiactivos S.A. (ENRESA), and ENUSA Industrias Avanzadas, SA SME (ENUSA) of Spain and Korea Radioactive Waste Agency (KORAD), Korea Atomic Energy Research Institute (KAERI), and Korea Electric Power Corporation Nuclear Fuel (KEPCO NF). The ENsa UNiversal (ENUN) 32P dual-purpose rail cask containing three surrogate PWR assemblies (the assemblies did not contain radioactive fuel) and 29 dummy assemblies (concrete masses) was instrumented with accelerometers and strain gauges. The basket, cask, cradle, and transportation platform were also instrumented. The accelerations and strains were measured during heavy-haul truck, ship, and rail transport, handling operations, and controlled rail tests at the Transportation Technology Center, Inc. (TTCI), a railroad testing and training facility in Pueblo, Colorado. During the test, 40 accelerometers, 37 strain gauges, and three Global Positioning System channels were used to collect 6 terabytes of data over the 54-day, 7-country, 12-state, and 8,500 miles of travel. While strains and accelerations have been measured on the exterior of transportation and storage containers, these measurements have never been collected on the fuel inside the container. The greatest strains and accelerations were observed during the testing at TTCI, specifically during the coupling test. Water transport strains and accelerations were the lowest and heavy haul and rail transport strains and accelerations were comparable. The handling tests were somewhat higher than the most extreme rail tests, except coupling. The observed strains were well below the yield points for spent nuclear fuel cladding demonstrating that the fuel can withstand the shocks and vibrations experienced during normal conditions of transport.

1. INTRODUCTION

The multi-modal spent nuclear fuel (SNF) transportation test was conducted in June–October 2017. The test was sponsored by the US Department of Energy (DOE). Two national laboratories (Sandia National Laboratories (SNL) and Pacific Northwest National Laboratory (PNNL)) participated in the design and implementation of the test. The international collaborators were Equipos Nucleares SA, SME (ENSA), Empresa Nacional de Residuos Radiactivos S.A. (ENRESA), and ENUSA Industrias Avanzadas, SA SME (ENUSA) of Spain and Korea Radioactive Waste Agency (KORAD), Korea Atomic Energy Research Institute (KAERI), and Korea Electric Power Corporation Nuclear Fuel (KEPCO NF).

Three 17 × 17 PWR surrogate assemblies were placed within the thirty-two cell ENsa UNiversal (ENUN) 32P dual-purpose rail cask basket along with twenty-nine dummy assemblies (concrete masses). The ENUN 32 P cask was provided by ENSA. One surrogate assembly was from SNL, one from Spain, and one from Korea. Selected rods within the PWR assemblies were instrumented with strain gauges and accelerometers. The ENSA ENUN 32P cask/cradle was placed, sequentially, on a heavy-haul truck, ships (coastal and transoceanic), and a railcar. The ENSA ENUN 32P cask, cask cradle, and transportation platforms (truck trailer, ship trailers, and railcar) were instrumented with accelerometers. During the test, 40 accelerometers, 37 strain gauges, and three Global Positioning System channels were used to collect 6 terabytes of data over the 54-day, 7-country, 12-state, and 8500 miles of travel. The processing and analysis of the data was performed in 2018. The main modes of transport are rail transport in the United States, truck transport in Spain, and ship transport in Korea, which served as a common denominator for the international cooperation among the three countries.

The test presented a unique opportunity to collect shock and vibration data for surrogate spent fuel assemblies in a full-scale transportation cask since data was collected for three different modes of transportation (heavy-haul truck, ship, and rail) and for intermodal transfer. Data was also collected during operations simulating the vertical placement of the ENSA ENUN 32P cask onto a surrogate storage pad. In addition, a series of short-duration controlled rail tests were performed at the Transportation Technology Center, Inc. (TTCI), a railroad testing and training facility in Pueblo, Colorado. The combination of different modes of transportation and handling offered an understanding of the cumulative effects of transportation and handling of SNF during normal conditions of transport.

2. TEST CONFIGURATION AND TRANSPORT ROUTES

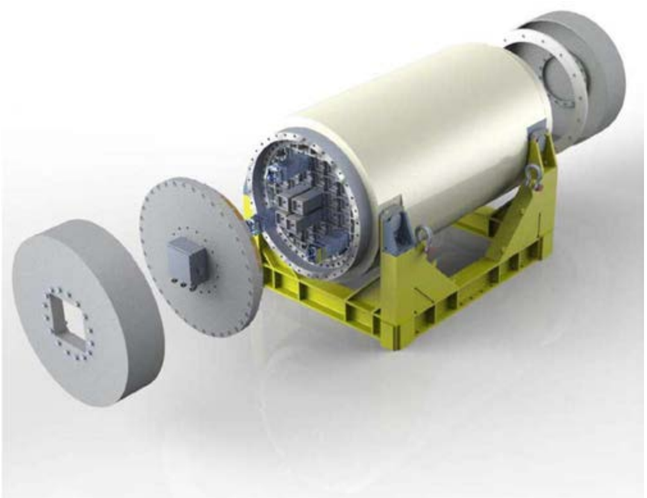
A diagram of the interior of the ENUN 32P basket, instrumentation lid, and the surrogate impact limiters is shown in Fig. 1. As configured for this test, the cask measured 5 metres in length with a body diameter of 2.65 metres. The loaded weight of the carbon steel cask was 120 tons and 137 tons with the surrogate impact limiters. Used to add the necessary weight, surrogate impact limiters were needed as real impact limiters would impede access to the cask for data collection.

The instrumentation of the surrogate assemblies is shown in Fig. 2. A total of 13 accelerometers and 37 strain gauges were installed on the assemblies and 6 accelerometers were installed on the basket (3 on the top and 3 on the bottom). The instrumentation of the exterior of the transportation system is shown in Fig. 3. A total of 21 accelerometers were installed on the transportation platform, cask, and cradle.

The data acquisition system and instrumentation were powered by twenty LifeLine Model GPL-8DL 12-volt batteries. Twenty batteries were sufficient to power the entire system for approximately three weeks.

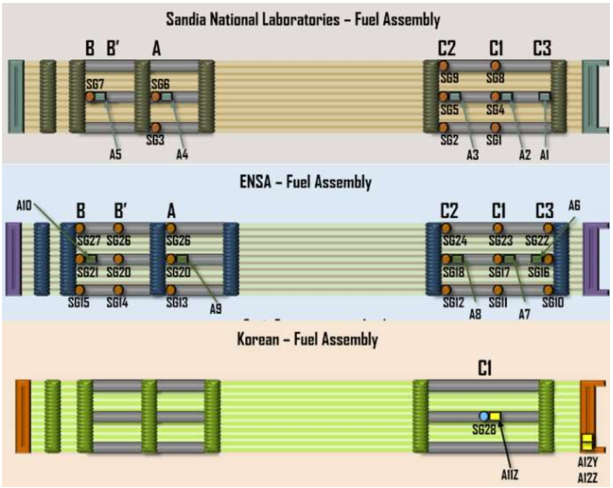
The first data collection took place during the dry storage handling simulation tests. These tests were conducted at ENSA's facilities in Maliaño, Spain. The data were also collected while the cask was loaded onto the heavy haul truck. Figure 4 shows the cask handling test.

The rail-cask was then transported by heavy-haul truck within northern Spain (the transport started and ended at ENSA's facility), by a smaller ship (coastal transport) from Port of Santander (Spain) to Port of Zeebrugge (Belgium), by a larger ship (trans ocean transport) from Port of Zeebrugge to Port of Baltimore), and by rail (round-trip from Baltimore to the TTCI near Pueblo, Colorado). A Kasgro KRL 370 12-axle heavy-duty rail flatcar was leased for rail transport. The transportation route is shown in Fig. 5. A number of short duration tests were conducted at the TTCI using the same railcar that transported the cask there. A short video documenting the major test events is available on YouTube [1].



Note: Surrogate assemblies are shown in blue and dummy assemblies are shown in gray. The ENSA assembly is on the top, the SNL assembly is at the bottom right and Korean assembly is at the bottom left.

FIG. 1. Schematic of the interior of the ENUN 32P basket, instrumentation lid, and the surrogate impact limiters.



Note: A is for accelerometers and SG is for strain gauges.

FIG. 2. Location and nomenclature of instruments on the fuel assemblies.

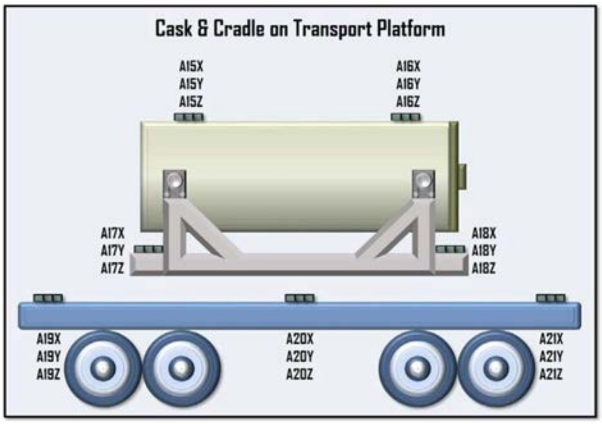


FIG. 3. Schematic of cask, cradle, and transportation platform accelerometer locations and nomenclature.

FIG. 4. Dry storage simulation test, ENSA's Facility in Maliaño, Spain.

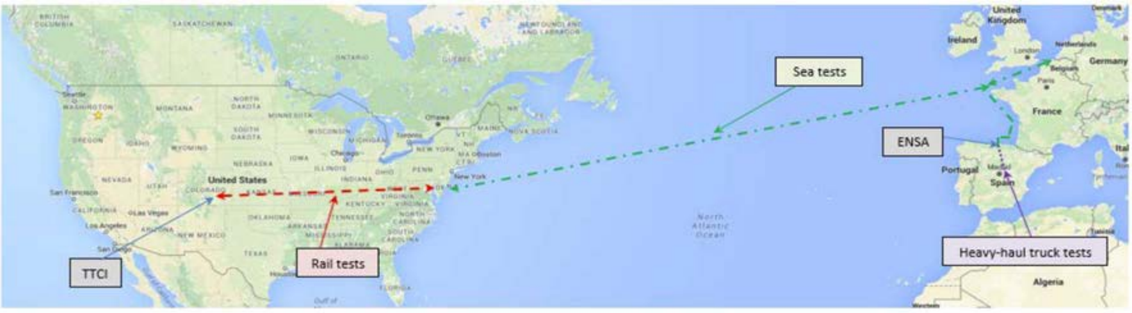


FIG. 5. Multi-modal transportation test route.

The test configuration used during the heavy-haul truck transport in Spain is shown in Fig. 6. Figure 7 shows how the cask was transported in the larger ocean ship. The test configuration was similar in the coastal transport, except the cask was on the same trailer as in the heavy-haul truck transport. Figure 8 shows the cask system on the Kasgro railcar. This configuration was used for the rail transport and in the TTCI tests.

FIG. 6. Heavy-haul truck transport in Spain.

FIG. 7. Cask lashed to the interior deck of the ocean ship "Tarago".

FIG. 8. Kasgro 12-axle railcar used for rail transport.

Table 1 provides the information on the distance and transport time for each transport mode. The time of transfer from one transportation mode to another is not included because no data were collected. The rail transport to Pueblo was via dedicated train and the travel time was much shorter. The rail transport from Pueblo was via general freight and the data collection stopped near St. Louis (half way to Baltimore) when the data acquisition system batteries reached its capacity.

Table 2 provides the description of the tests conducted at TTCI. The data frequency collection was 10 240 Hz during the TTCI tests and handling tests and 512 Hz during the other transportation.

TABLE 1. TRANSPORTATION ROUTE PARAMETERS

Transport Mode	Total Distance (mi)	Total Transport Time (hrs)
Heavy Haul	245	29
Coastal Freighter	929	120
Ocean Ship	4290	193
Rail from Baltimore (Rail1)	1950	59
Rail from Pueblo (Rail2)	1125	420
Total	8539	918

TABLE 2. RAIL TESTS PERFORMED AT THE TTCI

Test Description	Number of Tests
Twist and Roll	19
Pitch and Bounce	9
Dynamic Curve	24
Class 2 Rail Track (PCD)	17
Single Bump	8
Crossing Diamond	6
Hunting	23
Coupling Impact	10

3. SUMMARY OF THE TRANSPORTATION TEST ANALYSIS

Approximately 6 terabytes of data were collected during the multi-modal transportation test. All the data were analysed in order to envelop the responses of the different elements of the transportation systems, such as the cask, the cradle, the basket, and especially the surrogate fuel assemblies. The data were not filtered to assure that the resonance frequencies of all the elements of the transportation system would be captured. The data analysis included determining minimum and maximum accelerations/strains for each of 40 accelerometers and 37 strain gages for each TTCI and handling test and for each significant shock event during heavy-haul, ship, and rail transport. Google Earth was used to analyse the location at which the event took place. The results of the preliminary data analysis can be found in Ref. [2]. A complete analysis is documented in Ref. [3]. The following sections summarize some results of the analysis.

3.1. Cask handling operations

To obtain a useful representation of cask handling, a range of cask impacts were performed. Three ENSA crane operators raised and lowered the cask three times, where varying degrees of crane handling “aggressiveness” were used by each operator for their three respective tests. Figure 9 compares the maximum accelerations on the SNL assembly in dry storage handling tests and heavy-haul handling test. The heavy-haul handling test is very similar to the handling tests in Run 1 and Run 3 (first and second crane operators). The two handling tests, Drop 1 and Drop 2, in Run 5 (third crane operator) are significantly higher than all the other tests. The maximum strain observed on the SNL assembly was 40 micro strain (Drop 2, Run 5).

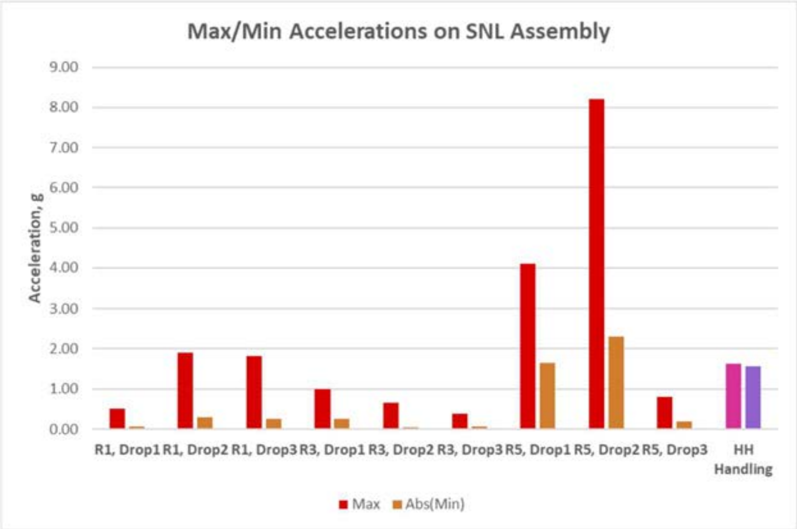


FIG. 9. Maximum accelerations on SNL assembly in Dry Storage Cask Handling Test and Heavy-Haul Cask Handling Test.

3.2. Heavy-haul transport

A total of 36 shock events were identified along the heavy-haul route, yielding one event per 6.8 mi. The majority of the events (78%) were caused by a vertical upset in the road (a bridge, crosswalk, a patchwork in asphalt, and imperfection in road surface). 11% of the events were associated with the turns. The remaining events did not have visible cause. These events did not cause substantial acceleration on either the transportation platform or on the SNL assembly. The maximum acceleration observed during the heavy-haul transport was related to travel over a bridge abutment. The maximum vertical acceleration on the back end of the transportation platform was 4.52 g. The maximum acceleration on the back of the SNL assembly was 0.52 g. The maximum strain on the back of the SNL assembly was 15.6 micro strain. Figure 10 shows strain time history of the SNL assembly during the maximum acceleration and strain event.

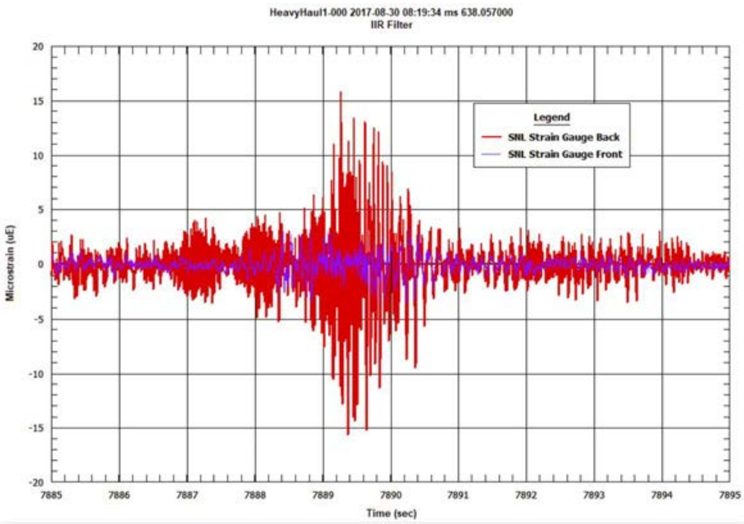


FIG. 10. Strain time history during maximum acceleration and strain event, heavy-haul transport.

3.3. Ship transport

The accelerations and strains observed during coastal freighter and ocean ship transport were very low. The accelerations observed were ≤0.3g (with a few exceptions) and the strains were ≤3 micro strain. The maximum acceleration on the transportation platform during ship transport was 0.38g. The maximum assembly acceleration was 0.12 g. The maximum strain on the SNL assembly was 3.15 micro strain.

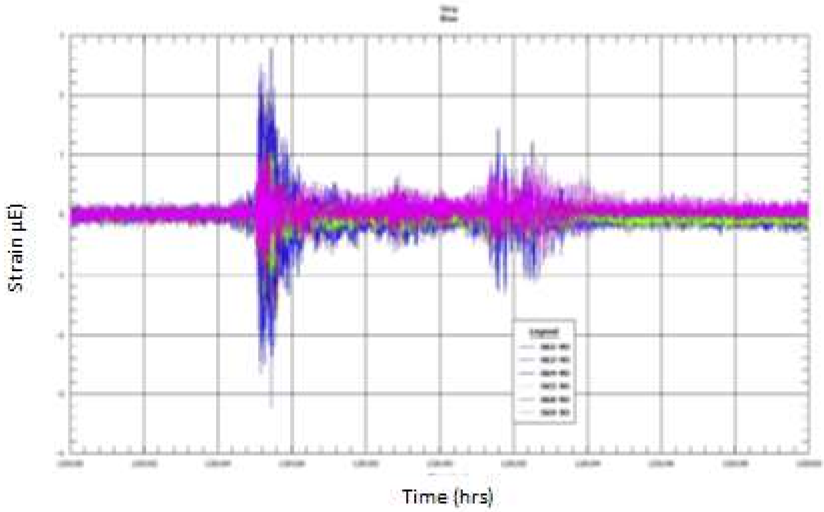


FIG. 11. Strain time history during maximum acceleration and strain event, ship transport.

3.4. Rail transport from Baltimore to Pueblo, Colorado (Rail1)

During the dedicated train transport from Baltimore to Pueblo (Colorado) the train travelled in the speed range 40–50 mph 23% of the time, 25–40 mph 68% of the time, 10–25 mph 8.8%, and <10 mph 0.2% of the time. A total of 2939 shock events were identified along the rail route — one shock event per 0.66 mi. The major events were track switches (629) and grade crossings (1029).

The maximum acceleration event in Rail 1 transport occurred over a diamond-crossing in Jacksonville, Illinois. The railcar was traveling approximately 36 mph. The absolute maximum peak acceleration was 8.68 g on the transportation platform. The maximum absolute assembly acceleration of 0.95 g was on the ENSA assembly. The maximum absolute strain was 20.7 micro strain on the SNL assembly front.

The maximum strain event occurred when the train passed over a switch in Kendall, Kansas. The railcar was traveling approximately 45 mph and experienced maximum absolute strain of 35.8 micro strain on SNL assembly. The absolute maximum accelerations were 3.78 g on transportation platform front, 0.66 g on the ENSA assembly front, and 0.63 g on the SNL assembly back end. The strain time history is shown in Fig. 12.

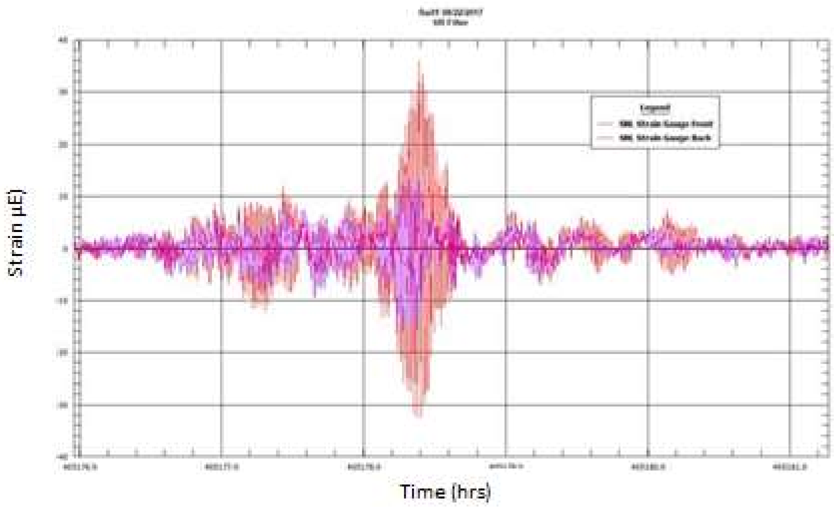


FIG. 12. Strain time history during maximum strain event, rail transport.

3.5. Rail tests at TTCI

A series of eight tests were performed at the TTCI. Each series included a number of tests conducted at different speeds to capture the test specific resonant speed. The TTCI tests were short duration tests with known conditions and with design parameters somewhat beyond the ones expected on the commercial railroads (track conditions, train speeds, and coupling velocities). These tests provided valuable insight into the response of the transportation system to the different types of transient inputs. Understanding of these responses was crucial for the analysis of rail, heavy-haul, and ship data.

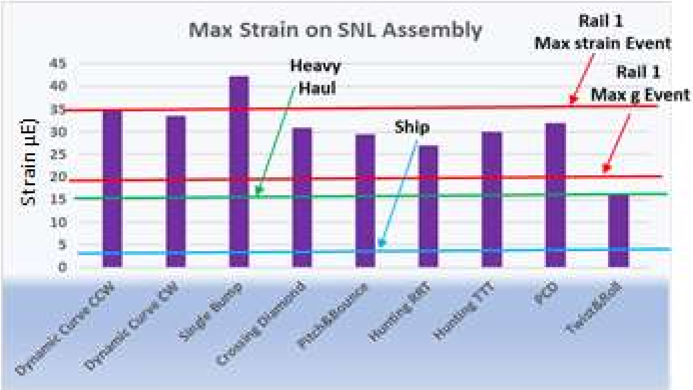


FIG. 13. Maximum strains on the SNL assembly in TTCI Tests compared to maximum strain in different modes of transport.

The TTCI tests with the highest accelerations and strains (except Coupling Impact Test) were: Single Bump Test, Pitch and Bounce Test, and Hunting on TTT Test. The Coupling Impact Test, particularly at high velocity, was the most severe event observed. Figure 13 compares maximum strains observed on the SNL assembly in the TTCI tests and different modes of transport. The tests at the TTCI bound the strains in rail, heavy-haul, and ship transport.

3.6. Rail transport from Pueblo, Colorado, to Baltimore

The rail transport from Pueblo (Colorado) to Baltimore (Rail2) on a regular freight train provided a valuable opportunity for analysing coupling events. Thirty coupling events were identified. Twenty-three events took place in major railyards and seven events took place in small railyards. A few coupling operations were performed at each railyard. The maximum acceleration observed on the SNL assembly was 1.05 g. The maximum strain was 38 micro strain. The maximum strain observed during coupling at TTCI was 99.0 micro strain in the 7.5 mph coupling.

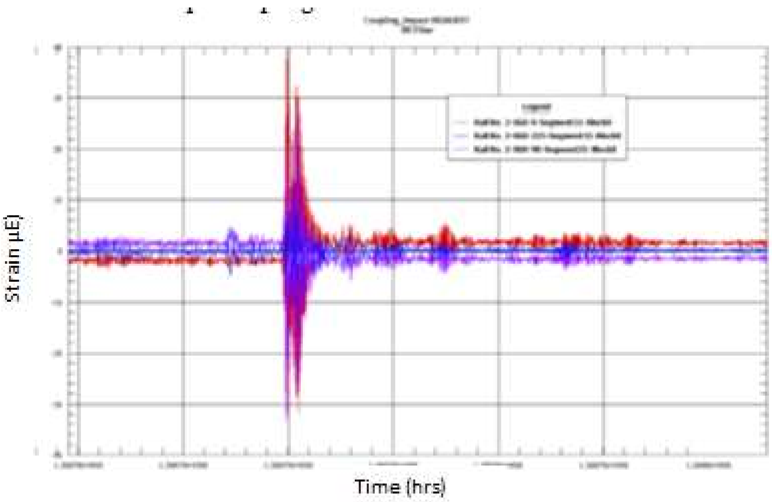


FIG. 14. Strain time histories on the SNL assembly front end, maximum amplitude coupling event during Rail2 transport.

3.7. Fatigue analysis

The strain data collected during the multi-modal transportation test were used to perform a fatigue analysis on the fuel cladding [4]. The ASTM Standard E1049 rainflow counting method was used to count the number of strain cycles in the data. Accumulated fatigue damage was calculated according to Miner's Rule, using an established irradiated zirconium alloy fatigue design curve [5]. Figure 15 shows the accumulated damage fractions for each strain gage for the Rail1 transport. A damage fraction of 1.0 indicates a fatigue failure, and accumulated damage in all cases is below 1E-10. This calculation method estimates that it would take 10 billion cross-country trips (2000 miles each) to challenge the fatigue strength of irradiated fuel cladding.

Another way to evaluate fatigue is to note that the maximum strain recorded during the multi-modal transportation test was less than 100 micro strain, and that strain is too small to cause any practical amount of fatigue damage to the material during a single trip (the yield strain for this material is greater than ~900 micro-strain). Using either analysis method, the conclusion is that cladding fatigue is not an issue during normal conditions of transport.

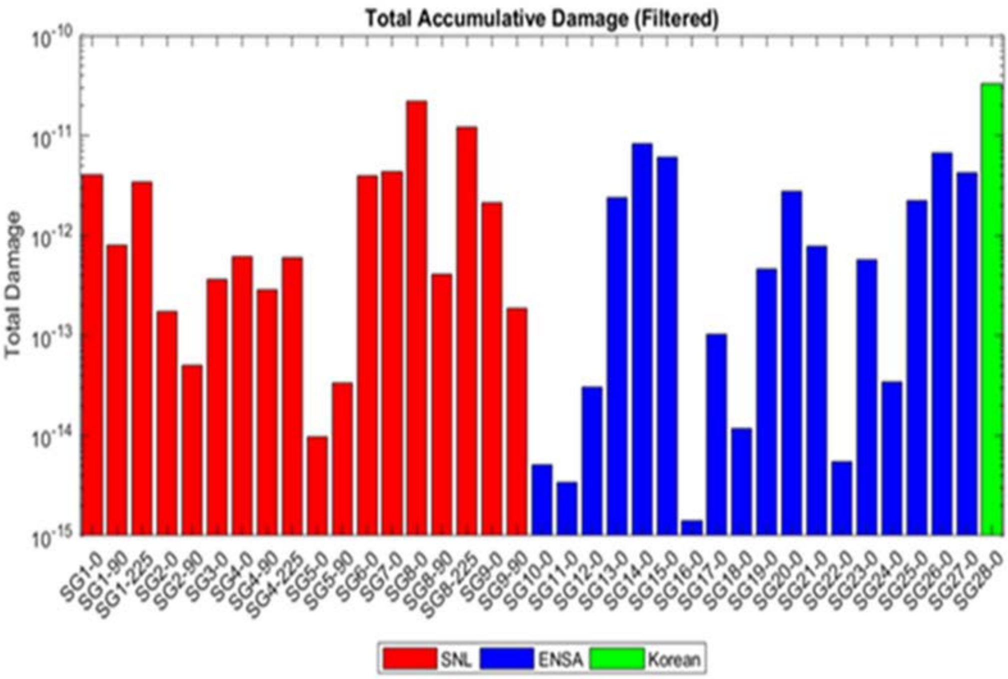


FIG. 15. Accumulated fatigue damage in Rail1 Transport (Baltimore, MD to Pueblo, CO).

ACKNOWLEDGEMENTS

The authors thank Ned Larson and John Orchard of DOE/LV for their support and advice for the cask transport tests. The authors would like to acknowledge the important contribution made by Paul McConnell, a principal investigator for the Cask Transport Tests Project, who has recently retired.

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Paper ID#205
EVOLUTION OF TRANSPORT REGULATIONS FOR SPENT FUEL

S. WHITTINGHAM
International Atomic Energy Agency
Vienna, Austria
Email: S.Whittingham@iaea.org

Abstract

The paper presents an overview of the evolution of the IAEA transport regulations for spent fuel since their inception in 1961. The challenges expected in the future for the transportation of spent fuel are highlighted. Besides compliance with the regulations, there is an additional need to assure for the 'transportability' of packages that are stored for extended periods before shipment for processing, conditioning, disposal of the spent fuel.

1. INTRODUCTION

The IAEA Regulatory Infrastructure and Transport Safety Section leads the development of IAEA Safety Standards related to regulating the safety of radiation sources and the safe transport of radioactive material. It supports Member States in their application of these Standards and of the Code of Conduct on the Safety and Security of Radioactive Sources.

The primary responsibility of the Transport Safety Unit is SSR-6 regulations which are adopted into the UN Model Regulations which in turn are adopted into the globally implemented International Dangerous Goods Regulations (IMDG Code) by the IMO[1] for transport by sea, and the globally implemented ICAO[2] Technical Instructions for transport by air. Land transport regulations are written and implemented by Member States. In 2018 IAEA issued a latest revision of the SSR-6 regulatory requirements which are currently in the process of being adopted by the UN Model Regulations, IMDG Code and ICAO Technical Instructions and Member States are aware of the appropriate actions regarding their land transport regulations.

2. THE HISTORY OF TRANSPORT REGULATION FOR SPENT FUEL

Since their inception in 1961, the transport regulation evolution can be illustrated by few examples, but the main question relates to the challenges in the future.

The Agency's Statute was approved on 26 October 1956 at an international conference held at United Nations headquarters, New York, and the Agency came into being when the Statute entered into force on 29 July 1957. The first session of the General Conference was held in Vienna, Austria, the permanent seat of the Agency, in October 1957.

In July 1959, the Economic and Social Council of the United Nations prepared a resolution entrusting the IAEA with the drafting of recommendations on the transport of radioactive substances. In May 1961, the Safety Series Number 6 (SS-6 regulations) on the safe transport of radioactive materials were produced. The timeline of these events demonstrates the importance of developing regulatory requirements for the safe transport of radioactive material

Section 15.3 of the Chapter on Fissile Materials introduces the general requirements for the carriage of fissile materials (including unirradiated and irradiated fuel) and further on, the section expands on the fissile materials in the form of nuclear fuel elements/ assemblies. Furthermore, in evaluating the inherent safety of the shipment of nuclear fuel elements, the following requirements were prescribed of which at least one had to be followed:

— The effective neutron multiplication factor (k_{eff}) of the system shall not exceed 0.9;

[1] International Maritime Organization
[2] International Civil Aviation Authority

- where mass is the controlling factor, the permissible value in any single package must not exceed 80% of the critical mass under the conditions of packing;
- if the geometry is the controlling factor, the permissible value for each controlling dimension must incorporate a 10% safety factor.

The IAEA publication on the regulations for the safe transport of radioactive materials has been revised several times since 1961 with its document number changed from Safety Series No. 6 to TS-R-1 and then SSR-6 to reflect the changes in document numbering systems used by the IAEA.

One of the major revisions was accomplished in 1964 when for the first time, the mechanical test requirements for Type B packages were introduced. In the first edition (1961), it was stipulated that a package must withstand maximum credible accidents, but no parameters were stated, and the concept was not very clear. Three years later in 1964, the test requirements that are intended to take into account a large range of accidents, which can expose packages to severe dynamic forces, although severity levels indicated by test criterion are not intended to represent a worst-case accident scenario.

Other examples of changes made in the IAEA regulations include:

- the introduction of the Criticality Safety Index (CSI) in the 1996 Edition;
- Type C package was introduced for air transport in 2005;
- Recently, in the 2018 Edition, ageing management became an explicit requirement for packages stored before shipment and the SCO-III category was introduced for the transport of very large decommissioning components that cannot be packaged.

Over the past 20 years there has been continuous increases in the computing power in terms of increased speed and modelling capabilities technical analyses codes. As a result, more parametric calculations in criticality assessment can be made with corresponding impact analyses to provide fuel assembly geometries and pin containment failure which can be used in the criticality assessments of the package design to ensure the package design remains sub-critical under normal transport and transport accident conditions. The objective of advanced computational analysis is to reduce uncertainties in the impact analyses and criticality assessments.

3. FUTURE OPPORTUNITIES AND CHALLENGES

Fuel designs for longer reactor cycle time and higher burnups is one of the objectives of the nuclear industry as these increase reactor performances by reducing outage time and improved utilization of the fuel. However, this will impose challenges for the transport of spent fuel in which it will be necessary:

- To demonstrate the structural integrity of the high-burnup fuel under prescribed transport impact accident conditions (geometry / pin containment);
- This data will be required for criticality assessment, which directly may affect the payload or indeed, in some instances, the fuel itself may not be transportable if the fuel assembly design suffers unacceptable failure of the fuel pins.
- Fuel designs (after irradiation) must be designed to withstand prescribed transport impact accidents; practical tests on spent fuel rods/ assemblies is not possible due to the lack of facilities to accommodate such testing. Still there is a need to know the mechanical properties the fuel assembly has after irradiation and what is the impact that causes decelerations of 120g, 130g or more depending on the package design.

These points illustrate that reactor operation is only one part of the nuclear fuel life cycle process and the controlling parameters will increasingly become transport regulatory requirements as the uncertainties of the structural performance of the spent fuel increases as a function of burnup.

Another challenge is the long term storage of spent fuel in Dual Purpose Casks (DPCs) which is gaining more prominence as an interim storage solution; reactor operation and storage of spent fuel are combined activities in most nuclear power programmes. The strategy to store spent fuel for decades until decisions on the next steps in the fuel cycle are made has storage time limitations and therefore the progressive increases in planned storage times originally ranging from 20–50 years is now cited in some discussions to be storage periods of 50–100 years.

The growing trend towards using the DPC for interim storage was recognized by the recent 2018 Edition of the SSR-6 regulation document which specifies that for packages designs that are intended to be transported after storage, there is now a requirement to have an ageing management system in place approved by the competent authorities as part of the Package design Safety Report upon which the package design approval certificate is based..

DPCs are often considered to solve the increased interim storage capacity and to provide a safe storage environment which can also be used for its future shipment. However, the challenges associated are multiple and have to be further addressed:

— It introduces a disconnect between an activity (transporting spent fuel) and its ongoing interaction with the public.
— As storing of spent fuel pending future decisions on the next step in the fuel management cycle becomes the norm, future generations will have little to no experience of transporting spent fuel in the public domain. The effects of this will be time-dependent. The longer the spent fuel is stored, the more influence the disconnect is likely to have.
— There is a need to develop a mechanism to evaluate 'transportability' which is more than solely compliance with the transport regulations. Transportability will remain unproven until preparations are being made for the first shipments of DPCs from storage in the future.
— Transportability may in itself provide a limit to the interim storage period
— One alternative that may provide longer interim storage periods may be to use a shielded container that is subsequently loaded into a transport cask that is designed, approved and manufactured nearer the time of the intended shipment programme of the shielded containers from the storage facility.

The further in the future we plan for subsequent shipments of long term stored DPCs, more consideration should be given to the wider issues that may influence the ability to complete those shipments; wider than regulatory compliance. It is important to include such considerations in the strategic planning to minimize the possibility of significant costs, delays, uncertainties and loss of credibility could be incurred to complete what would then be a near term planned shipment programme.

Paper ID#196
INTERNATIONAL SHIPMENTS OF SENSITIVE NUCLEAR MATERIALS
Experience of International Nuclear Services in safe and secure transport of sensitive nuclear materials across international borders

M. CROWTHER
International Nuclear Services
United Kingdom
Email: mark.crowther@innuserv.com

Abstract

This paper explains the key activities that International Nuclear Services undertakes to consistently deliver high quality and reliable international marine transports of Category I to III nuclear materials.

1. INTRODUCTION

International Nuclear Services (INS) and its majority-owned subsidiary Pacific Nuclear Transport Ltd. (PNTL) have been carrying out transports of different categories of nuclear materials for well over forty years, travelling over five million miles without any nuclear or security incident.

This record has been achieved through significant and sustained investment in the capability of assets, people and systems by INS and its international partners and stakeholders. INS has applied its unique experience in this field to ensure a rigorous and uncompromising approach to regulatory compliance, safety, security, capability and communication.

2. INS AND PNTL

INS is a subsidiary of the Nuclear Decommissioning Authority (NDA), a UK public body responsible for the safe and efficient cleanup of the UK's nuclear legacy. INS's role is to support delivery of the NDA strategy by providing specialist nuclear transport, design and licensing services. A significant part of INS's work involves the transportation of spent fuels, highly active waste and special nuclear material. INS is the only organisation in the world that offers a high-security Category I nuclear shipping capability and therefore plays a key role in delivery of UK and global security and non-proliferation goals. Since becoming an NDA subsidiary in 2008, after the restructuring of the UK civil nuclear industry, INS has undertaken many complex nuclear transport projects for various countries including Japan, USA, Sweden, Italy, Belgium, France, Switzerland and Germany. INS has provided end-to-end solutions for transport of such materials as plutonium, highly enriched uranium, MOX fuel, vitrified high level waste and spent fuel.

INS is the majority owner of PNTL, the world's most experienced marine shipper of sensitive nuclear materials and is responsible for its management and operations. PNTL was formed in 1975 to provide a strategic transport solution for transports from Japan to Europe and back. The experienced PNTL crew are highly trained in the specialised skills needed to transport nuclear materials.

INS and PNTL's people are experts in engineering, package design and licensing, nuclear transport operations, project management, international nuclear law, shipping, stakeholder relations, security and resilience, health and safety, emergency response, contract management and commercial services.

Of the four vessels operated by INS, NDA owns one vessel and PNTL three vessels, all INF3 class ships are specifically designed to transport up to Category I material safely and securely. All the ships are crewed by PNTL.

3. REGULATORY FRAMEWORK

Regulation of safe and secure nuclear transport is comprehensive and had been developed since the 1960s through the leadership of IAEA and its member states. There is a robust framework of international legal and regulatory requirements and standards that apply to safety and security in the multi-modal approaches to transport of nuclear materials.

International regulations include (not exhaustive):

— IAEA Regulations for the Safe Transport of Radioactive Material (SSR-6).
— UN Recommendations on the Transport of Dangerous Goods.
— IMO International Maritime Dangerous Goods Code (IMDG Code).
— International Code for the Safe Carriage of Packaged Nuclear Fuel, Plutonium and High Level Radioactive Wastes on Board Ships (INF Code).
— European Agreement concerning the International Carriage of Dangerous Goods by Road (ADR).
— IMO International Convention for the Safety of Life at Sea (SOLAS).
— IAEA Planning and Preparing for Emergency Response to Transport Accidents Involving Radioactive Material Safety Guide.
— IMO International Convention for the Prevention of Pollution from Ships (MARPOL).
— IMO International Safety Management Code (ISM Code).
— United Nations Convention on the Law of the Sea (UNCLOS).
— ICAO Technical Instructions for the Safer Transport of Dangerous Goods by Air.
— IATA Dangerous Goods Regulations.
— OTIF Regulations Concerning the International Carriage of Dangerous Goods by Rail (RID).
— UN Convention on the Physical Protection of Nuclear Material (CPPNM).
— IAEA Nuclear Security Recommendations on Physical Protection of Nuclear Material and Nuclear Facilities (INFCIRC/225/Revision 5).
— IAEA Security in the Transport of radioactive Material (Nuclear Security Series No. 9).
— IAEA Security of Nuclear Material in Transport (Nuclear Security Series No 26-G).
— IMO International Ship and Port Facility Security Code (ISPS Code).

INS and PNTL are required to follow these regulations and, as UK companies operating from within the UK, all of the UK regulatory requirements. For transport across and into other countries, INS and PNTL must also meet all of the legal and regulatory requirements of those countries.

An effective regulatory framework is central to not only maintaining the safety and security of nuclear transports but also gaining the understanding and support of the many national and international stakeholders who maintain an interest in INS's activities.

INS is one of the founder members of the World Nuclear Transport Institute (WNTI) and is active in supporting WNTI in its work with key intergovernmental organisations in promoting an efficient, harmonised international transport safety regime. WNTI is a non-governmental organisation that has consultative status with the IMO and observer status with the IAEA. In addition, WNTI has consultative status with the United Nations Committee of Experts on the Transport of Dangerous Goods, Category B Liaison Membership with the International Organization for Standardization (ISO), and information status with the American National Standards Institute (ANSI) N14 Committee.

4. ELEMENTS OF EFFECTIVE NUCLEAR MATERIALS TRANSPORT

Through working closely with partner organisations, INS offers complete solutions for transport of nuclear materials from door-to-door across the world. INS transports over the past forty years have covered over five million miles by sea, as well as rail and road. The company has carried out movements of over 2000 transport flasks, 20 shipments of high level vitrified waste residue, 12 shipments of MOX fuel and various other significant shipments in support of governments' efforts to reduce the global threat associated with certain nuclear materials. Whilst INS has maintained a flawless radiological safety record throughout that period, it has continued to develop and improve its approach in order to adapt to changing circumstances and the resulting regulatory and, best practice landscape.

The key to INS's success has been to develop with its partners an uncompromising approach that focusses on key areas including planning, safety, security, reliability, compliance and expertise.

4.1. Safety in depth

4.1.1. Design and Licensing

Shipping assets, the vessels and transport packaging or flasks, are specifically designed for the purpose in hand.

4.1.2. Package Design and Licensing

IAEA Regulations for the Safe Transport of Radioactive Material set the basic requirement that safety is vested in the package in order to provide all the levels of protection needed for the transport. The package must be appropriately shielded to provide protection from radiation to workers and the public; protect from the effects of an accident or fire; prevent potential dispersion of contents.

Regulations cover five classifications of package (Excepted, Industrial, Type A, Type B, Type C), each providing standards based on levels of radioactive activity and the form of the material. Test requirements are set for each package depending on risks that may arise during the transport.

Type B packages are needed to transport material with higher levels of radioactive activity and are used frequently by INS in transports around the world.

In addition to standard design requirements for all packages, Type B are required to withstand:

- Drop test from 9 metres;
- Puncture test;
- Fire (at least 800ºC for 30 minutes);
- Immersion (up to 200 m for 1 hour).

The INS flask design and licensing team are experienced in finite element analysis, shielding, criticality assessments and mechanical design, allowing them to design and license new packages, as well as applying their expertise to existing packages for transport of varying materials. INS can self-certify packages up to Type A classification. The team can design complete end-to-end transport solutions, including engineering, bespoke transport packages and bring the benefits of its expert ongoing contribution to IAEA regulations to ensure efficient and compliant transports.

4.1.3. Vessel Design and Licensing

In 1993, the International Maritime Organisation (IMO) established the International Code for the Safe Carriage of Packaged Irradiated Nuclear Fuel, Plutonium and High Level Radioactive Waste on Board Ships (INF Code). The INF Code became mandatory in 2001 and sets stringent standards for the three classes of INF ships, described in Table 1.

TABLE 1. CLASSES OF INF SHIPS

INF Class 1 Ship	INF Class 2 Ship	INF Class 3 Ship
Ships which are certified to carry materials with an aggregate radioactivity less than 4000 TBq	Ships which are certified to carry irradiated nuclear fuel or high level radioactive wastes with an aggregate radioactivity less than 2 x10^6 TBq and ships which are certified to carry plutonium with an aggregate radioactivity less than 2 x 10^5 TBq	Ships which are certified to carry irradiated nuclear fuel or high level radioactive wastes, and ships which are certified to carry plutonium with no restriction on the aggregate radioactivity of the materials

Source: WNTI

The four vessels that INS operates, Oceanic Pintail MV, Pacific Heron MV, Pacific Egret, and Pacific Grebe were designed to meet the requirements for INF3 vessels. The INF code was based upon the first generation of PNTL ship design and INS was operating vessels to INF3 standards twenty years before IMO made INF Code mandatory.

INS's design of the three PNTL vessels is an evolution of the first generation of INF3 class ships and exceeds the requirements of the INF3 code. The cargo compartments are protected by a double hull and all essential systems on the ships are duplicated and separated to provide high reliability and accident survivability. If any important system fails during a voyage, either owing to mechanical or system failure or as a result of an accident, there is always a second system ready to be brought into operation. In addition, no tanks or spaces containing oils or other pollutants are positioned directly adjacent to the outer hull to minimise the chances of pollution should the outer hull be ruptured during an incident.

In summary the key features are:

— Double hull throughout, with additional strengthening surrounding the holds;
— Independent engines, machinery and steering gear;
— Auxiliary generators;
— Duplicated and multiple systems;
— Hold cooling plant located outside holds for easier maintenance;
— Integrated bridge system;
— No oil tanks adjacent to outer hull;
— Security features incorporated into design;
— Enhanced security features;
— Improved environmental and safety performance;
— Advanced fire detection and firefighting systems.

Some of these features are illustrated in the image of the Pacific Heron in Fig. 1.

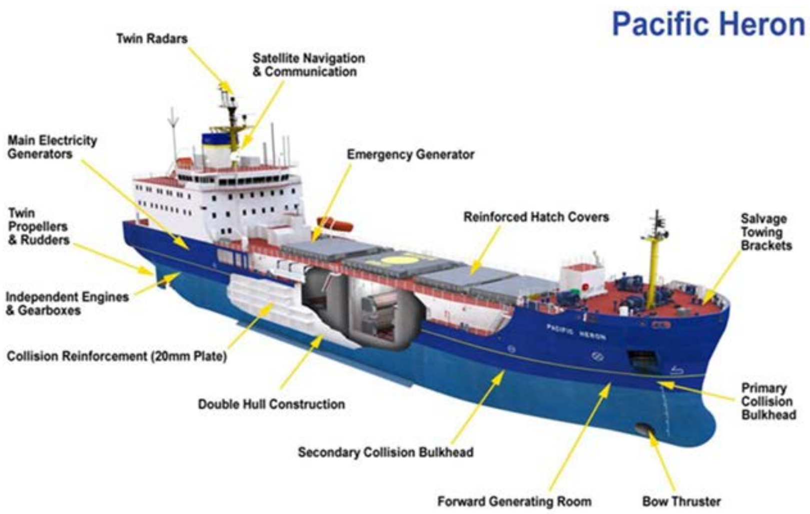

FIG. 1. Cutaway image of pacific heron.

4.2. Assurance - Planning, Preparation and Implementation

INS plans each of its shipments meticulously. For more complicated transports, planning may need 12 months and sometimes longer. Ahead of shipment planning, INS will also undertake feasibility studies as necessary to ensure that licensing and all the options for delivery are fully understood and developed.

For each individual transport, INS assures itself that all applicable requirements of the transport regulations and all additional assurance requirements are met. INS has a robust Transport of Nuclear Cargoes process (extract shown below in Fig. 2) that supports the operations and approval teams in ensuring these requirements. To verify that all requirements are in place, INS also implements a decision matrix under its Nuclear Transport Safety Committee Review process to reviewing key nuclear safety aspects of transports, including all transport modes and multi-modal transfer points.

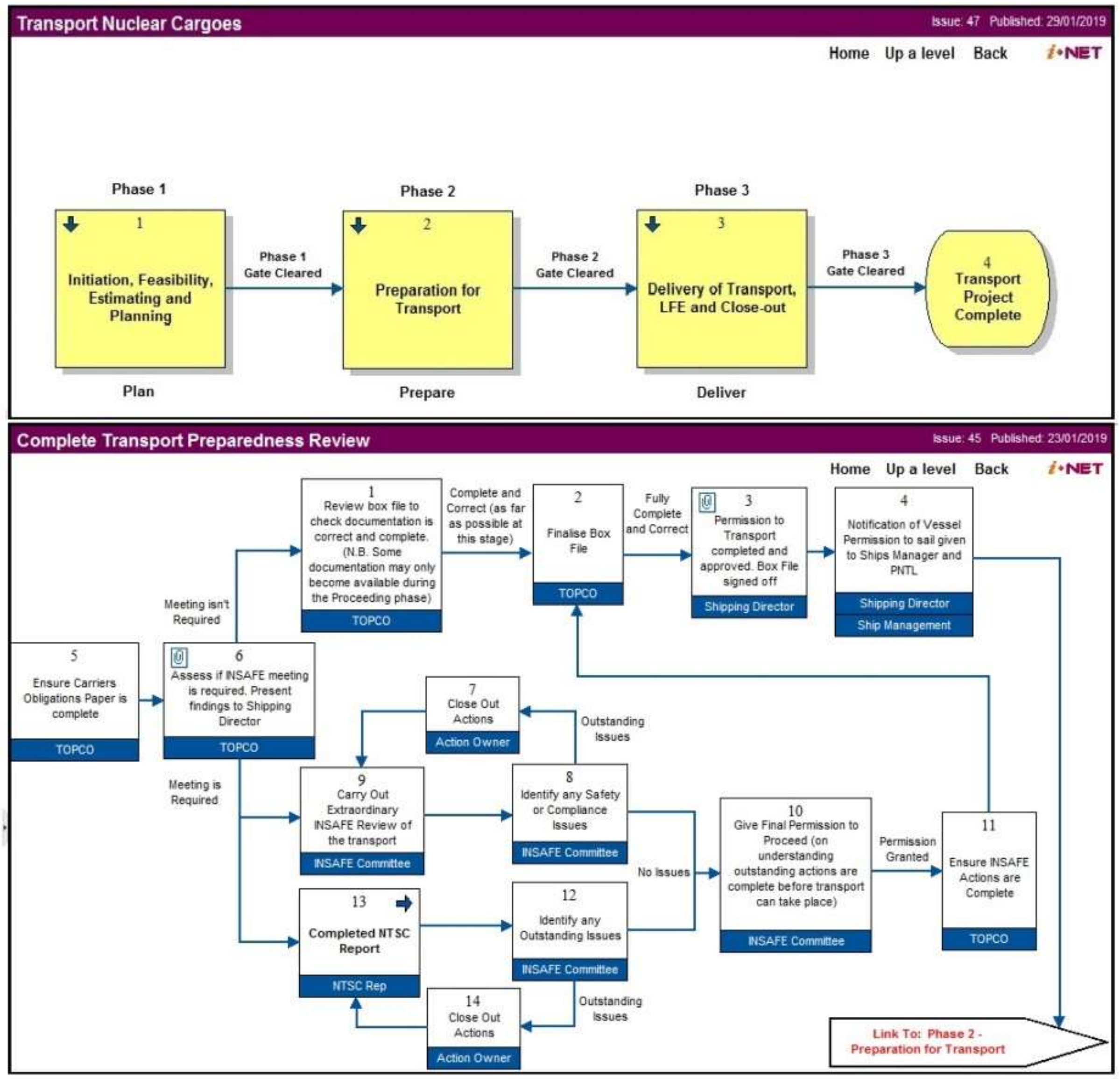

FIG. 2. Example of nuclear cargoes transport process.

The summary of both reviews is then presented and discussed at a Readiness Review Meeting, convened to confirm that all requirements are in place and the transport can proceed. The Readiness Review Meeting also ensures that any learning taken from previous shipments has been implemented for the upcoming shipment.

This whole process is subject to external audit and regular self-verifications to ensure that the process remains fit-for-purpose.

4.3. Security

INS and its shipping subsidiary PNTL are subject to UK security regulations. CPPNM and IAEA security requirements flow through to the UK in the Nuclear Industries Security Regulations 2003 (NISR2003) and the Security Assessment Principles for the Civil Nuclear Industry (SyAPs), regulated by the Office of Nuclear Regulation. INS is the duty holder for provision of security for nuclear transports in line with UK regulations. It is responsible for providing a Transport Security Statement laying out INS's overall approach to meeting all required security regulations, as well as a Transport Security Plan for each individual shipment.

INS vessels have enhanced security features and for higher category shipments are armed to protect the vessels against potential security threats. The UK Civil Nuclear Constabulary (CNC) is responsible in the UK for the physical protection of civil nuclear sites and facilities from potential threats. INS partners with CNC and their specialised Strategic Escort Group to provide security services for transports. Requirements will vary depending on the nature of the shipment.

INS implements a comprehensive security programme to ensure that everyone involved in transport of nuclear materials is an active participant in a strong security culture. INS approach includes:

- Leadership and management: ensuring clear governance, leadership by example, policies and roles & responsibilities;
- Security culture: a programme to educate, enable, encourage, evaluate and continually improve security culture;
- Competence: a programme to ensure appropriate training and qualifications of responsible personnel;
- Supply chain management: setting expectations of supply chain in managing sensitive information and assets;
- Cyber and information security: protection of systems and Sensitive Nuclear Information;
- Reliability, resilience, sustainability: accreditation, examination, maintenance, testing, sustainability;
- Physical protection systems: protecting against theft, sabotage, design, vulnerable assets;
- Emergency response: exercising and testing arrangements;
- Policing/guarding: ensuring an effective relationship with CNC, local police and security guards;
- Vetting: ensuring a reliable and trustworthy workforce through national vetting, aftercare and a programme of insider threat measures.

4.4. Emergency Response and Transport Control

INS has a proven and well exercised 24-hour emergency response and transport monitoring capability. Any issues that arise during transports are managed through this capability.

In support, INS has established a network of global response partners; including marine salvage, air transport and health physics. It has put in place dedicated tactical and strategic UK command centres for transports and has highly trained and experienced personnel in command, control and communication. INS has worldwide access to emergency response equipment including pre-determined grab-bags positioned globally.

The response capability, systems and procedures are tested regularly including desktop, simulated and real-time multi-disciplinary exercises.

4.5. Information and Communication

During transport operations INS follows a general principle of transparency, at least to the extent permitted by security considerations. An open and honest approach to information disclosure helps to establish credibility and confidence. INS also proactively engages with the public and communities in advance of transport operations. It works closely with its partners to explain its transport activities, in particular with its local stakeholders in its home port of Barrow-in-Furness, and with the countries and organisations along the routes travelled by INS vessels. These relationships are key to maintaining confidence in INS's ability to transport materials safely and securely.

5. CONCLUSION

Transport of nuclear materials is an essential part of civil nuclear industry operations, allowing for movement of fresh fuel, spent fuel, waste and other products. The basis for the safe and secure transport of sensitive nuclear materials is the sound international regulatory framework, stretching across a wide range of disciplines, such as safety, security, maritime, rail, road, air. In addition to the key standards set by regulation, INS has developed systems and capability to set the bar for delivery of nuclear transports to a high level, resulting in no nuclear incidents over more than forty years of operation. Key to delivering these high standards is the integrity of transport assets, integrity of management systems, competence of staff in planning, delivery and security, constant challenge to prevent complacency and ensuring effective relationships with national and international stakeholders.

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3.4. TRACK 4 – RECYCLING AS A SPENT FUEL MANAGEMENT OPTION

Overview prepared by A. Khaperskaya (Russian Federation) and H. Zaccai (World Nuclear Association), **Track Leaders**

Part of the presentations were focused on the development of existing mature technology for reprocessing. The French La Hague reprocessing plant and the Russian reprocessing plant (RT-1) PO "Mayak" enhancing the range of LWR and Russian Reactors used fuels, including damaged fuels, to be treated and performing investments to both increase competitiveness and secure long term operations. This is leading to continuous development and implementation of new technologies. In the La Hague plant these improvements include the implementation of a Cold Crucible Induction Melter, R&D performed regarding clogging issues in dissolution and separation steps, corrosion issues in evaporators and vitrification steps in reprocessing technological scheme. The Russian reprocessing plant RT-1 plans for increasing the capacity of reprocessing from 400 to 600 tHM/year by 2022, with upgrading the technology, including new cutting machine and a voloxidation unit for tritium removal from effluent releases.

Another part of the presentations was focused on the status of existing technology of Pu mono-recycling in MOX fuel (experience of MOX fuel supply and MOX fuel performance enhancement) and the development of technologies of multi recycling in the existing fleet of LWR (in Russian – several type of REMIX fuel for multi recycling of RepU and Pu, in France-CORAIL and MIX fuel for Pu multi recycling). These recycling options can reduce the need for natural uranium from 25% to 35% compared to the open fuel cycle, and can be a sustainable solution for the transitioning period from once-through recycling (current solution) to a closed fuel cycle with FR. The preliminary assessments of the MIX and CORAIL concepts show they are capable of recycling spent MOX and ERU fuel and of stabilizing the spent fuel and plutonium inventories. Russian studies showed the economic benefits for REMIX technology in the closed fuel cycle in comparison with the open fuel cycle.

The Japanese presentation was focused on the benefits of recycling from the point of view of minimizing the amount of waste as a target for environmental load reduction of the geological disposal for high level radioactive waste with discussion of the impact of SF characteristics (fuel burnup, spent fuel cooling period) and radioactive waste characteristics (waste loading in vitrified waste, separation of minor actinides as heat-generating nuclides).

Presentations from Russia were also connected with approaches to evaluate spent nuclear fuel reprocessing products activity and volume equivalence which is to be returned to the spent fuel supplier state after reprocessing. Another theme of discussion was provided from the perspective of the nuclear regulator - the evaluation of the safe conditions of chemical processes of nuclear fuel cycle facilities.

The 5 posters were presented: 2 from Egypt, 1 from Turkey, 1 from Russian Federation, and 1 from Syrian Arab Republic. The posters focused on different approaches to recycling technology (Russian Federation, Turkey, Egypt and Syrian Arab Republic), and the calculation of the inventory and activity of the radionuclides in the spent fuel from the reactor core of Fukushima Daiichi (Egypt).

Session 4.1: Recycling as a spent fuel management option

Session Chairs: C. Roussel (France) and A. Kirkin (Russian Federation)

Session 4.1 comprised of five presentations, two from France, two from Russian Federation and one from Japan.

- **Paper ID#127 by B. Morel (France)** presented an overview of the solutions brought by R&D programs to maintain a high level of reliability in La Hague reprocessing plant. Removing clogging from dissolution equipment, columns of extraction have been done thanks to basic rinsing operations. To avoid corrosion issues, metallic materials are carefully selected. Corrosion issues bring the development of tele manipulator very resistant to radiation to repair equipment in very hostile environment and of the cold crucible melter to vitrify the fission product solution rich in molybdenum.
- **Paper ID#53** ***(Invited)*** **by A. Sheremetyev (Russian Federation)** presented the experience of Mayak Production Association. Various spent nuclear fuel can be treated thanks to the adaptation of facilities for VVER-1000, fast reactors spent nuclear fuel, or spent nuclear fuel from research reactor facilities or propulsion facilities and defective spent nuclear fuel. According to the author the line will be equipped with a heavy-duty cutting machine and a voloxidation unit that allows containing tritium.
- **Paper ID#33 by H. Asano (Japan)** presented the Japanese research programme which aims to optimize the fuel cycle conditions regarding the environmental load reduction of the geological disposal for high level radioactive waste. Fuel burnup, spent fuel cooling period, waste loading of vitrified waste and separation of minor actinides have been studied and compared to the present conditions and their impact on geological repository.
- **Paper ID#126 by P. Breitenstein (France)** presented a solution developed to manage damaged fuel assemblies: packaging, transport logistics, wet or dry storage, and reprocessing. These damaged fuel assemblies are from different types of reactors and from various countries. Various cask designs exist, and specific operations are developed to manage defective fuels to decrease global risk.
- **Paper ID#19 by A. Rodin (Russian Federation)** presented an approach based on an adaptation of existing methodology of explosion safety assessment considering the specificity of nuclear industry. The method could allow researchers to evaluate the safe conditions of chemical processes of nuclear fuel cycle facilities and it can be used as a basis for further safety requirements development.

Session 4.2: Recycling as a spent fuel management option

Session Chairs: A. Khaperskaya (Russian Federation) and Y. Guoan (China)

Session 4.2. comprised of five presentations, three from France and two from Russian Federation.

- **Paper ID#72** ***(Invited)*** **by C. Delafoy (France)** presented the French experience in using MOX fuel in LWR and new approaches to Pu multi-recycling strategies in LWRs with the CORAIL and MIX fuel assembly. The paper also contents the adaptations to be implemented at the MELOX production plant to face the degradation of the Pu isotopic vector of MOX fuel and its higher Pu content. The CHROMOX product which involves Cr_2O_3 doping is characterized by an enhanced homogeneity of the Pu distribution in the fuel and an increased matrix grain size. With these evolutions, larger internal pressure margins are anticipated as well as some enhancement in the retention of gaseous fission products in accidental conditions by reduction of restructured areas

- **Paper ID#55 by A. Grol (Russia)** presented the technical and economic parameters of REMIX fuel performance in the LWR fuel cycle with describing five options of REMIX-fuel technology for Pu multiple recycling in existing fleet LWR. It was highlighted that economic assessment has demonstrated economic benefits for REMIX technology in the closed fuel cycle in comparing with the open fuel cycle.
- **Paper ID#92 by E. Buravand (France)** presented an oxidizing digestion process applied to Sodium-Cooled Fast Reactor (SFR) MOX fuel recycling to recover plutonium and reduce solid residue volumes. The higher plutonium contents in SFR MOX fuels require specific treatment steps to prevent plutonium retention in residues at the front end of the reprocessing cycle. It has been demonstrated that the nature and the composition (U, Pu) of the dissolution residues are heavily dependent on the initial fuel material and on the irradiation conditions in the reactor. The oxidizing digestion of dissolution residues permits the recovery of up to 99% of residual plutonium. The efficiency of oxidizing digestion was seen not only for plutonium recovery but also for metallic element dissolution.
- **Paper ID#94 by C. Chabert (France)** presented several multiple recycling options for plutonium fast reactors or PWRs for the transitioning period from once-through recycling (current solution in France) to a closed fuel cycle with fast reactors. The preliminary assessments of the MIX and CORAIL concepts show they are capable of recycling spent MOX and ERU fuel and of stabilizing the spent fuel and plutonium inventories. It was highlighted that compared with the once-through recycling of uranium and plutonium, these concepts reduce the need for natural uranium over 25% compared with an open fuel cycle.
- **Paper ID#10 *(Young Generation Challenge Winner)* by A. Kirkin (Russian Federation)** presented approaches to evaluation of spent fuel reprocessing products activity and volume equivalence to be returned to a supplier state after the reprocessing in the Russian Federation. The current approaches in the Russian Federation for the reprocessing products returning to the Supplier's state correspond to international approaches and are based on the analysis of dose coefficients of various nuclides. The authors proposed another approach with including dose coefficients from uranium-plutonium fuel (REMIX - or MOX-fuel) and reduced volume of returned radioactive waste in the form of cesium and strontium fraction in borosilicate matrix.

The five posters were presented: 2 from Egypt, 1 from Turkey, 1 from Russian Federation, 1 from Syrian Arab Republic.

- **Paper ID#29 by M. Shaat (Egypt**) presented the calculation and simulation of the inventory and activity of the radionuclides in the spent fuel of the reactor core of Fukushima Daiichi, Unit 1 Nuclear power plant Accident, using a Monte Carlo transport Code, Monte Carlo N-Particle eXtended (MCNPX) for the purpose of future reprocessing of the spent fuel and used in manufacturing of MOX fuel.
- **Paper ID#13 by B. B. Acar (Turkey)** presented the result of the scenario investigation based on the storage of both U+Pu and fission products (FP) + minor actinides (MA) of the complete co-processing of spent fuel from a typical LWR. It is underling that complete co-processing of spent fuel as a back end nuclear fuel cycle strategy can be a valuable option as a waste management strategy. The potential product (U+Pu) and waste (FP+MA) were separated and can be further used.

- **Paper ID#16 by N. Mohamed (Egypt)** presented a new design of a PWR fuel assembly for direct recycling of the PWR spent fuel in CANDU reactors. Recycling directly the spent PWR fuel in CANDU reactors will increase the fuel burnup, will degrade the plutonium vector which enhances the proliferation resistance and the produced plutonium per unit energy would be reduced.
- **Paper ID#59 by N. Kovalev (Russian Federation)** presented a status of the REMIX fuel concept development in Russia with the results of irradiation and post-irradiation study. It is underlying that REMIX fuel can provide multiplied recycling in existing fleet of LWR with 100% core loading.
- **Paper ID#155 by S. Dawahra (Syrian Arab Republic)** presented the result of calculation of the possibility of extending MOX fuel share in commercial nuclear power plants, including VVER-1000 and RBMK-1000, using different ways as: reducing the burnup, reducing the time of the MOX assemblies irradiation in the reactor, increasing the fraction of MOX assemblies in the core, and reducing the plutonium enrichment in the MOX fuel.

Paper ID#53
EXPERIENCE AND PROSPECTS OF SPENT NUCLEAR FUEL REPROCESSING AT MAYAK

A. SHEREMETYEV, D. KOULUPAEV
Federal State Unitary Enterprise Mayak Production Association
Ozersk, Chelyabinsk Region, Russian Federation
Email: AVSheremetjev@po-mayak.ru

Abstract

Development of atomic power engineering on a global scale has made it necessary to address problems associated with spent nuclear fuel (SNF) management. Operation of nuclear facilities resulted in accumulation of large amount of SNF with various compositions and geometries. The SNF is accumulated both during electric power generation at NPPs and operation of naval propulsion reactors by surface and submarine fleet, and during research and development of new approaches to fuel management carried out at nuclear research centers.

There are two competing approaches to management of generated SNF. One approach is based on long term storage with subsequent direct disposal (open cycle), while the other one is connected with radiochemical reprocessing (closed cycle). The Russian Federation adopted a strategy of closed nuclear cycle with SNF reprocessing and recycling of recovered products resulted from reprocessing. Implementation of this strategy is closely connected with SNF reprocessing carried out by Mayak PA.

1. INTRODUCTION

Mayak Production Association is the first industrial facility of the Russian nuclear industry. It was established to produce fissile plutonium-239. June 19, 1948 is considered to be the date of Mayak PA foundation. On this date the first nuclear reactor in the Eurasian continent reached its 100 MW designed capacity.

Radiochemical technology of plutonium extraction from the irradiated uranium became the most challenging and hazardous part of the Uranium Project. December 22, 1948 is considered to be the date when a radiochemical plant was put into operation. On this date the first batch of irradiated slugs was delivered to the radiochemical plant from the large-scale nuclear reactor.

The radiochemical plant has always been and is still an important part of an integral process flow and organizational structure of Mayak Production Association.

In spite of evident success made by the radiochemical plant at that time, it came to face new, more ambitious challenges of improving technology for weapons-grade plutonium extraction and increasing its production output. Thus, on July 25, 1953 the decision was made to construct a backup plant, which was an analogue of the radiochemical plant already in operation.

The backup plant that was commissioned in 1959 ensured systematic increase in plutonium production at operating process lines. The required amounts of nuclear material were provided in full measure. On October 1, 1971 the first and the second plant lines were integrated into a single subdivision of Mayak Chemical Combine, i.e. into a radiochemical plant (the RT-1 Plant) that started adopting SNF reprocessing technology.

On December 30, 1971 the first special train delivered spent fuel assemblies from the MR reactor of I.V. Kurchatov Institute of Atomic Energy to the fuel storage pool of the RT-1 Plant. The first process line for spent nuclear fuel fragmentation and dissolution was put into operation on March 29, 1977.

Since then more than 6300 tHM (including foreign fuel) have been delivered to the Plant for radiochemical reprocessing.

RT-1 Plant is the only SNF reprocessing plant in Russia focused on acceptance, interim storage and reprocessing of various types of SNF. The Plant is also involved in management of generated radioactive waste of any level of activity. Initially the Plant was designed for reprocessing SNF from NPPs with VVER-210, 360, 440, RBMK-1000, BN-350 and BN-600 power reactors, from research reactors of Russian and foreign research centers and from naval propulsion facilities of submarine and surface fleet. And the RT-1 Plant demonstrates sustainable development as it evolves continually and expands its technological capabilities. Today the RT-1 Plant is a versatile multiproduct facility. Current technological capabilities of transporting and reprocessing SNF, as well as handling radioactive waste are unprecedented across the globe.

Many years of SNF shipment have provided Mayak PA with vast experience in the field of transportation. Mayak PA is experienced in using all modes of transportation including railway, motor-vehicle, water and air transport. Today air transportation of irradiated research fuel is unprecedented practice in the world.

2. EXPERIENCE IN NPP SNF REPROCESSING

2.1. VVER-440

For a long time the major SNF reprocessed at the RT-1 Plant has been the SNF from VVER-440 reactors most extensively used in the European part of the Russian Federation and spread around Eastern Europe countries. After interim storage in reactor cooling ponds (3 to 5 years as a rule) SNF from VVER-440 reactors is still transported from both Russian and foreign NPPs to Mayak PA for reprocessing. The total amount of SNF transported and reprocessed over the entire history of the Plant is more than 5300 tHM.

2.2. VVER-1000

Power-generating units with VVER-1000 and VVER-1200 reactor facilities succeed the fleet of VVER-440 reactors that are gradually being decommissioned. Today 12 power units of this type are operated in the Russian Federation; 8 more power units are under construction.

In 2015 and 2016 Mayak PA implemented activities on preparation of the facilities for acceptance and reprocessing of SNF from VVER-1000 reactors for the purpose of enhancing cost-effectiveness of the reprocessing facility and increasing the capacity rate of the RT-1 Plant.

In the framework of this Project the RT-1 Plant developed and implemented a new transportation flow chart for VVER-1000 SNF handling supported by nuclear and radiation safety analysis and execution of all necessary permitting documents. Cold tests were performed that used a shipping cask for VVER-1000 SNF and a full-size SFA simulator.

Pilot hot shipment of SNF from the Rostov NPP was carried out in December 2016. The first 12 SFAs from VVER-1000 were accepted and reprocessed at the radiochemical plant of Mayak PA using a new transportation flow chart.

The results of this work enabled Mayak PA to transport to the radiochemical plant and to reprocess SNF from VVER-1000 reactors, including experimental, defective and leaking SFAs on regular basis from 2017. Reprocessing of SNF from VVER-1000 and involvement of uranium and plutonium thus recovered into production of nuclear fuel for NPPs will ultimately provide closure of the nuclear fuel cycle for VVER-1000.

2.3. Fast Reactor SNF

The history of reprocessing of this type of fuel at the RT-1 Plant goes as far back as 1983, when the Plant started reprocessing SNF of BN-350 reactor (now located in the territory of the Republic of Kazakhstan). After the USSR breakup, shipments of BN-350 SNF from Kazakhstan to Mayak PA were stopped.

Later on when BN-600 reactor facility was commissioned at the Beloyarsk NPP, reprocessing of the fuel from this reactor was initiated. From then on, reprocessing of SNF from the BN series reactors has become a routine process for the Plant; one of three process lines of the RT-1 Plant is dedicated especially for handling SNF of this type. The total amount of such fuel reprocessed at the Plant is more than 500 tHM.

In 2018 BN-800 reactor facility that is the next reactor facility of this series, was commissioned at the Beloyarsk NPP. Taking into account experimental nature of fast neutron reactors and plans to convert the core of the BN-800 reactor facility to 100% loading with mixed uranium-plutonium fuel, Mayak PA started preparing for MOX-fuel reprocessing. The first experimental reprocessing of MOX-fuel from BN-600 reactor (8 SFAs) was carried out in 2012 and 2013. It demonstrated technical readiness of the facility for handling mixed uranium-plutonium fuel. The first shipment of SNF, including the MOX-fuel, from the BN-800 reactor for radiochemical reprocessing will take place in 2020.

3. EXPERIENCE IN REPROCESSING SNF FROM RESEARCH AND NAVAL PROPULSION FACILITIES

3.1. SNF from Research Reactor Facilities

The fuel from the research reactors and facilities is characterized by a wide variety of both geometry of the fuel assemblies being used and the corresponding fuel compositions.

20 research reactors are currently in operation in Russia. The research reactor SNF is reprocessed at the RT-1 Plant, Mayak PA, after interim storage at the research center sites.

Versatility of radiochemical reprocessing of SNF implemented at Mayak PA makes it possible to reprocess (recycle) the majority of currently used fuel assemblies of nuclear research facilities. It is worth noting that several significant projects aimed at expanding the range of SNF types suitable for reprocessing have been recently implemented at the RT-1 Plant. Reprocessing of such fuel compositions as U-Mo, U-C, U-N, Umet and of such a complex composition as U-Be, was initiated between 2011 and 2016.

The project focused on adopting a technology for electrochemical dissolution of fuel was completed at the RT-1 Plant in 2018. This is one of the key points that will allow the Plant to become a multi-fuel facility capable of reprocessing all possible (currently in use) fuel compositions including such a complex one, as uranium-zirconium fuel. The first experimental operations on dissolution of uranium-zirconium SFAs from the research reactor at A.I. Leipunsky Institute of Physics and Power Engineering were conducted in 2019. In the future the electrochemical dissolution technology can be used to solve the problem of handling corium (damaged fuel) of the Fukushima Daiichi NPP, Japan, and to reprocess plutonium metal and its alloys.

The promising project that can be implemented at Mayak PA in the near future is reprocessing SNF with uranium-thorium fuel composition from the Elk River NPP, USA, currently stored in Italy. Feasibility study performed at Mayak PA demonstrated that this fuel can be reprocessed based on the process scheme currently in use at the RT-1 Plant without major upgrade of the production facility and deterioration in the end product quality. Taking into account considerable amount of SNF with uranium-thorium fuel composition mainly from research reactor facilities accumulated all over the world, the potential for its reprocessing at Mayak PA is significant and prospective.

3.2. SNF from Naval Propulsion Facilities

Russia is operating five icebreakers including Taymyr (1988), Sovetskiy Soyuz (1989), Vaygach (1990), Yamal (1992), 50 Let Pobedy (2007) and one icebreaking LASH-carrier Sevmorput (1988).

Interim storage of ice-breaker fleet SNF is performed at floating maintenance bases and onshore storage facilities of FSUE Atomflot. Part of SNF of the ice-breaker fleet is unloaded from the storage facilities of nuclear ice-breakers by the floating maintenance base (Lotta vessel) and is stored in TUK-120 casks at Atomflot storage site. Later on the ice-breaker fleet SNF will be transported on regular basis for reprocessing at Mayak PA by railroad in special vehicles.

4. EXPERIENCE IN FOREIGN SNF SHIPMENT

Beyond the boundaries of Russia, there are NPPs with VVER-440 and VVER-1000 reactor facilities constructed from Russian designs and research reactors that use nuclear fuel of Russian origin.

History of international shipments to Mayak PA goes as far back as 1971 when the RT-1 Plant was established. It was at that time when VVER-210 fuel from the GDR was delivered to the Plant.

As far as NPP fuel is concerned, since then shipments from 8 countries, such as Hungary, Finland, Germany, Bulgaria, Czechia, Slovakia, Armenia and Ukraine have been performed. More than 2400 t of foreign fuel was reprocessed using capacities of the RT-1 Plant.

KS-150 SNF from the Slovak NPP is currently stored at Mayak PA and will be reprocessed at Cutting and Canning Department upon completion of reprocessing SNF from AMB and EGP-6.

Since 2006 high-enriched SNF from foreign research reactors of Russian origin from 13 countries has been delivered to Mayak PA in the framework of the Russian-American RRRFR Programme (Russian Research Reactor Fuel Return). For this purpose, multimodal transport scheme has been implemented. The joint international project has made it possible to remove high-enriched uranium potentially suitable for the manufacture of nuclear weapons from more than ten countries. Each shipment was unique in its kind. It posed challenging technical problems and required overcoming difficulties associated with SNF import into the Russian Federation. Implementation of international contracts for return and reprocessing of spent nuclear fuel from foreign research institutes allowed the region to fulfill large-scale environmental programs.

In the framework of a separate project in September 2015 liquid SNF based on aqueous solution of uranyl-sulphate enriched to 90% U-235 was imported from the IIN-3M research reactor in Uzbekistan to the territory of Russia. Preparatory work included development of a technology and a set of equipment for discharge of the liquid SNF from the reactor into the interim storage tanks, the SNF reloading into the shipping cans, as well as equipment for loading of canned liquid SNF into SKODA VPVR/M cask using a transfer cask. Besides, a technology and special equipment were developed to receive cans with liquid SNF at Mayak PA. The batch of liquid SNF (about 27 litres) in TUK-145/C (Type C package that includes SKODA VPVR/M cask and a shock-absorber) was delivered by the AN-124-100 aircraft to Russia for reprocessing at the radiochemical plant, Mayak PA.

5. NUCLEAR LEGACY PROBLEMS

5.1. Experience in Defective SNF Handling

Another indicator of the RT-1 Plant versatility is its capability of handling defective irradiated fuel assemblies.

Mayak PA took part in several projects, including the international ones, connected with transportation and reprocessing of defective SFAs. One of the first projects implemented in 2010 was focused on transportation of leaking fuel of the RA research reactor from Serbia and Montenegro (Vinča Nuclear Institute). Later on, the defective fuel assemblies from the Paks NPP, Hungary (2014) and Kozloduy NPP, Bulgaria (2016) were delivered to Mayak PA.

Significant efforts have been made by Mayak PA to prepare for handling defective and out-of-specification fuel of the RBMK-1000 reactors.

The RBMK-1000 power reactors are operated at the Leningrad NPP, the Kursk NPP and the Smolensk NPP located in the European part of Russia. Annually about 3500 SFAs are unloaded from the RBMK-1000 reactors. Then the SFAs are stored in the reactor storage pools and stand-alone on-site wet storage facilities.

Since 2012 activities have been underway to switch over to a safer dry method of the RBMK-1000 SNF storage at the Central Storage Facility at Mining and Chemical Combine in Zheleznogorsk (Krasnoyarsk region). In compliance with the approved procedure, sealed SFAs without major rack defects (i.e. on-specification SFAs) are subject to dry storage at on-site storage facilities and transportation to the Central Storage Facility. Dry storage of defective and leaking SFAs (out-of-specification SFAs) at on-site storage facilities, their transportation and acceptance at the Central Storage Facility were not provided for. Therefore, as an alternative option of handling out-of-specification SFAs from the RBMK-1000, their reprocessing at Mayak PA was proposed.

With the aim of validating and testing engineering solutions, in 2011 a pilot batch of leaking SNF was delivered for reprocessing from Leningrad NPP power unit No.2 using well-proven and simple to operate TUK-11 casks. Preliminary disintegration of SFAs into fuel bundles was carried out in the hot cell of power unit No.2. In 2014 a batch of sealed out-of-specification SNF (22.4 t) was transported for reprocessing in metal concrete TUK-109 casks from the central spent nuclear fuel storage facility of the Leningrad NPP. Disintegration of SFAs into fuel bundles was carried out in the hot cell of the central SNF storage facility. Thus, depending on the SFA condition and capabilities of the facilities, several logistics options for the RBMK-1000 SNF transport were implemented.

Since 2015 the RBMK-1000 SNF has been transported to Mayak PA in the framework of the federal target programme in the amount of up to 30 tons per year. Uranium recovered in the course of the RBMK-1000 SNF reprocessing is used efficiently in combination with recovered uranium resulting from reprocessing of other SNF types for the purpose of product batching and producing commercial batches of uranyl nitrate hexahydrate with specified parameters of enrichment and uranium-232 isotope content.

Thus, out-of-specification and defective RBMK-1000 fuel is regularly delivered to the RT-1 Plant and reprocessed according to the standard transport flow chart. The main condition to be fulfilled is placing defective SFAs into sealed package (can) before their dispatch to Mayak PA.

5.2. Removal of SNF from Naval Onshore Bases

On June 7, 2017 the first SNF batch from Andreev Bay (Murmansk region) was transported on board Rossita container ship for subsequent reprocessing at the RT-1 Plant. The first batch included 350 SFAs, each of which contained up to 20 kilograms of SNF. It should be noted that a total of 22 000 fuel assemblies are stored in the storage facility, which corresponds to the contents of 100 naval nuclear propulsion reactors. Shipment of the first SNF batch is an example of successful multilateral international cooperation aimed at solving challenging problems of nuclear legacy in the North-West of Russia, enhancing nuclear and radiation safety and improving the environmental situation.

SNF and radioactive waste handling facilities, as well as engineering systems were established in the framework of the federal target programme Industrial Disposal of Weapons and Military Equipment from 2011 to 2015 and for the Period until 2020 in cooperation with governments of Great Britain, Norway, Sweden, Italy, as well as with European Commission and Northern Dimension Environmental Partnership (contributors to the NDEP Support Fund are the European Union, Belgium, Great Britain, Germany, Denmark, Canada, Netherlands, Norway, Russia, Finland, France and Sweden; the Fund manager is the European Bank for Reconstruction and Development). The total budget for construction of infrastructure and improvement of radiation environment at Andreev Bay for the period starting from 1999 exceeded 8 billion rubles. The established SNF handling infrastructure will help reduce the time of SNF removal by more than a factor of three.

5.3. AMB Reactors

One of the most pressing problems in the field of nuclear and radiation safety is management of SNF from AMB reactors. Two AMB reactors of the Beloyarsk NPP were shut down in 1989. SNF was unloaded from the reactors, and it is currently stored in cooling ponds at the Beloyarsk NPP and at 'wet' storage facility at Mayak.

Spent fuel assemblies from AMB reactors have the following defining features: about 40 types of fuel compositions and large dimensions (SFA length makes approximately 13 m). The main problem of storing the assemblies at the Beloyarsk NPP is corrosion of basket-tube holders and the cooling pond linings.

Range of works concerning management of SNF from AMB reactors is provided with a view to reprocessing the fuel at Mayak. As of today, methods and process regulations of radiochemical reprocessing of SNF from AMB reactors were chosen and substantiated. In 2011 pilot reprocessing of fuel from AM reactor (analogue of SNF from AMB reactor) was performed. Construction of Cutting and Canning Department is underway.

At the same time, a process flowchart of shipment of SNF from AMB reactors was established. The flowchart includes a range of technical activities and arrangements at Mayak and the Beloyarsk NPP, as well as works on forming a special train with unique TUK-84 transportation packages for delivery of SNF from AMB reactors. As a result, a holder with SNF assemblies from AMB reactors, subject to intermediate storage at Mayak, was shipped in November 2016.

Today the project is of significant importance in the field of nuclear and radiation safety in the atomic sector, because it enables reprocessing of SNF from the first and the second shut-down units of the Beloyarsk NPP (a total of 122 holders). Rosenergoatom JSC in its turn will be able to begin their decommissioning. Establishing a system for management of SNF from AMB reactors will give Mayak an opportunity to begin accepting on regular basis the SNF using the adopted transport flowchart, and to remove the whole stock of the fuel from AMB reactors from the Beloyarsk NPP, which will meet the most important challenge in the field of nuclear and radiation safety.

The beginning of cutting and reprocessing of SNF from AMB reactors is scheduled for 2024. SNF stored at the storage facility at Mayak PA should be reprocessed in the first place, then prospectively the entire amount of fuel from the Beloyarsk NPP pools will be delivered at Mayak for reprocessing.

5.4. EGP-6

At the moment there is no decision concerning the back end of the single type of SNF, i.e. fuel from EGP-6 reactors (the Bilibino NPP). This is a long fuel assembly (as SNF from AMB reactors), its fuel composition is similar to the composition of one of AMB fuel versions, that is why reprocessing of this type of SNF will be possible at Mayak after Cutting and Canning Department starts its operation. However, the project implementation demands extremely high expenses because of the great distance to the Bilibino NPP, lack of infrastructure for retrieval of SNF and its further shipment from the NPP site, and necessary transport infrastructure around the site.

6. PROSPECTS AND NEW PROJECTS

Considering very significant amount of SNF from VVER-1000 accumulated in the territory of the Russian Federation, as well as the rate of construction progress for NPPs with reactors of this type, both in RF and abroad, the RT-1 Plant focuses in the first place at increasing the amount of fuel reprocessed for SNF of this type. With this in view, upgrading the second process line is scheduled. The line will be equipped with a heavy-duty cutting machine (as in the 3d line) and a voloxidation unit that allows containing tritium and solving the problem of tritium condensate discharge into the storage ponds.

Commissioning of the second fast neutron reactor (BN-800) in Russia and a prospect of constructing units with BN-1200 reactors make it necessary to increase capacity of reprocessing SNF from fast neutron reactors, including MOX-fuel. The current capacity of the RT-1 Plant can make 400 tHM/year. Completion of the RT-1 Plant upgrading is scheduled for 2022. The upgrading will enable increase in amount of reprocessed SNF from thermal reactors to 600 tHM and over, including possibility to reprocess fuel from at least three units with fast neutron reactors.

RT-1 Plant will be technically capable of reprocessing necessary amount of any SNF type, including defective and out-of-specification ones. Thus, the Plant will completely meet requirements and fulfil tasks of potential customers.

Upgrading the 3d line enabled reprocessing of SNF of foreign design. In 2017 Mayak successfully implemented the project on demonstration reprocessing of a mock-up of TVS-Kvadrat (square SFA), i.e. a complete analogue of foreign assemblies from PWR reactors.

7. CONCLUSION

It is worth mentioning the following RT-1 achievements of recent years:

- equipment of the reprocessing facilities at RT-1 Plant was upgraded, which enabled expansion of the range of the fuel reprocessed (RBMK-1000, BN-MOX, VVER-1000, both standard and in baskets) and significant increase in uranium product output;
- SNF from atomic submarines is shipped from Andreev Bay on regular basis, the RT-1 Plant will become the culminating point in the process of solving the problem of nuclear legacy of Northwest Russia;
- technology of reprocessing SNF with uranium-beryllium, uranium-carbide, uranium-molybdenum fuel compositions was adopted;
- technical capability of reprocessing SFA of foreign design was demonstrated;
- technology of reprocessing SNF with uranium-zirconium fuel composition was developed and adopted;
- technical capability of reprocessing SNF with uranium-thorium fuel composition was confirmed.

For the period from 2025 to 2030 the main vectors of the RT-1 development are:

- increasing rates of reprocessing to 600 tHM and over;
- implementing HLW partitioning methods;
- arranging integrated production for U-Pu fuel fabrication.

Even now current capabilities of the RT-1 Plant provide solutions of the most complicated technical problems concerning closure of the nuclear fuel cycle, thus enabling increase in nuclear and radiation safety level at nuclear facilities.

Paper ID#72
PLUTONIUM RECYCLING THROUGH LWR MOX FUEL: TODAY AND TOMORROW

C. DELAFOY[a], J. JONNET[a], Y. RUGAMA[b]
Framatome
[a] Lyon, France
[b] Paris, France
Email: christine.delafoy@framatome.com

V. GARAT
Orano Cycle MELOX
Bagnols sur Cèze, France

Abstract

Recycling operations have been mastered for long in France, from the plutonium separation to the irradiation of MOX fuel, as France committed itself towards recycling plutonium in PWRs since 1987. Today, the French reactors using MOX are operated according to fuel management allowing equivalent performance of energy supplied with the same reliability as those using UO_2 fuels.

The paper first presents the experience feedback obtained up to 65 GWd/tHM (rod average). Fuel microstructural evolutions under operations as well as the behavior of fission products have been thoroughly examined. A somewhat higher fission gas release is observed compared to UO_2 fuel mainly due to the higher power levels of the MOX fuel and its more heterogeneous microstructure. To keep the parity with UO_2 in the future, MOX evolution based on advanced microstructures is considered to provide the required performance. In that respect, the CHROMOX microstructure obtained by Cr_2O_3 doping shows an enhanced homogeneity notably with smaller primary blend agglomerates and increased matrix grain size. With these evolutions, internal pressure margins are anticipated and better retention of gaseous fission products in accidental conditions by reduction of restructured areas.

To sustain the use of MOX fuel in the future, the second part of the paper presents the adaptations to be implemented at the MELOX production plant to face the inherent degradation of the Pu isotopic vector of MOX fuel and its higher Pu content from increased core management cycle length.

In addition, Pu multi-recycling strategies in LWRs are studied with new fuel technologies. In order to be able to use low quality Pu in a PWR spectrum, fissile uranium needs to be added. With the CORAIL-A option, developed by Framatome and Orano, the assembly contains about half of MOX fuel rods and the remaining as UO_2 rods. By contrast, the MIX fuel assembly contains only MOX rods with an enriched uranium matrix that compensates the Pu degradation. Development of those fuel technologies, that could be coupled with the most advanced Framatome fuel assembly design GAIA, will offer flexibility to switch to future technically and economically robust advanced cycles in current or future LWRs with a limited impact to the reactor design and its performance. These developments will allow implementing efficient solutions bridging the gap with the potential development of GEN IV reactors.

1. INTRODUCTION

Plutonium recycling through MOX fabrication and irradiation in Light Water Reactors (LWRs) is a proven industrial solution with both fuel design and supply chain well mastered. The R&D efforts have allowed enhancing the performance and meeting utilities requirements, i.e. MOX parity with UO2 fuel in reactors in a complete safety approach, especially at high burnup.

Though the situation is highly satisfactory, especially in France, Framatome and Orano are both committed to improve the MOX fuel product technology. A special development effort was invested to change the MOX fuel pellet microstructure for enabling a better distribution of the plutonium rich-phases within the fuel matrix. This evolution is desired to account for plutonium isotopic quality in the future leading to increase the plutonium content of the MOX fuel for LWRs. In that case, margins to safety criteria will decrease, notably fuel rod internal pressure margins which may question the parity principle. To maintain that objective, doping MOX fuel with chromium oxide is considered with the so-called CHROMOX concept. Besides favoring the homogenization of uranium and plutonium, doping also activates the grain growth. The first results obtained after irradiation in an experimental reactor are on line with the expectations, namely enhanced fission gas retention in the fuel matrix, a beneficial feature for both normal operation and accidental conditions.

Mono-recycling in LWRs is an interesting strategy to reach a rapid decrease of the total used fuel inventory resulting in a reliable and safe solution. It also maximizes the energy that can be extracted from uranium resources. However, a plutonium multi-recycling strategy in LWRs can be also considered to go further in the use of energy potential contained in both UOX and MOX fuels. In this context, the CORAIL-A and MIX fuel assembly designs have been developed with special care to the minimization of power distributions inside the assembly and of interactions with the neighbors.

These different aspects are addressed in greater details hereafter, as well as the necessary adaptations to be implemented at the MELOX production plant to face the MOX fuel developments considered in the future.

2. MOX FUEL EXPERIENCE FEEDBACK AND MID-TERM EVOLUTION

2.1. MOX fuel performance in LWRs operation conditions

MOX fuel is used in Light Water Reactors (LWRs) since 1972 with no safety related issues. At mid-2018, Framatome had delivered about 8700 fuel assemblies in more than 40 reactors worldwide. Today in France, 'MOX Parity' is achieved with UO_2 fuel. The AFA 3GA MOX product has a reference assembly Pu content of 8.65% (9.77% for the high content zone) and is irradiated in an annual quarter core reload basis up to a maximum burnup of 52 GWd/tHM (corresponding to 59 GWd/tHM maximum fuel rod burnup).

The qualification of the so-called Parity MOX 52 product is based on a large-scale R&D programme carried out since the beginning of Pu recycling in LWRs. The database includes several surveillance and analytical programs, in and out of pile, performed within the French cooperation scope with CEA, EDF and Framatome or within international programs. In pile analytical experiments have been performed in normal and off-normal conditions up to high burnup with the aim of collecting relevant data to develop and validate behavior models. Currently, the experience feedback reached a maximum fuel rod average burnup of 65 GWd/tHM.

MOX fuel features and evolutions during irradiation have been thoroughly examined. It is concluded that most of the physical properties of MOX fuel for LWR applications do not differ significantly from those of UO_2 fuel because of the relatively low Pu content. However, noticeable exceptions exist which directly affect MOX fuel in-pile behavior:

- Higher fuel temperatures are observed due to lower thermal conductivity and higher reactivity for MOX fuel [1];
- Higher rod internal pressure at end of life is observed as a consequence of Fission Gas Release (FGR) which is somewhat larger compared to UO_2. This behaviour results mainly from the linear heat rates experienced by MOX fuel during their last irradiation cycles [2]. In current French MOX fuel managements, FGR is especially sensitive to power levels and fuel temperatures during the 3rd irradiation cycle at a stage where incubation of the insoluble fission gases in the fuel matrix has been long enough to reach a saturation threshold (so-called Vitanza threshold for FGR >1% [3]).
- In addition, helium production during irradiation and its release contributes to the rod internal pressure increase especially at high burnup [4]. The different sources of helium production in MOX fuel under irradiation are α decay from specific minor actinides, ternary fissions and $^{16}O(n,\alpha)$ reactions.
- Increase in MOX fuel creep rate and plasticity flow leading to a better pellet-clad-interaction resistance relatively to UO_2 fuel [5].

Beyond its properties, MOX fuel differentiates from UO_2 by its microstructure when manufactured with the MIMAS (MIcronized MASterblend) process as is the case at the Orano/ MELOX facility. This process guaranties a homogeneous Pu distribution at the pellet scale, but the MIMAS MOX pellets have a multi-phasic microstructure at the microscopic scale [2]. The microstructure consists of three phases: Pu-rich agglomerates with the Pu content of the master blend, i.e. ~ 28 %, U-rich agglomerates with a very low Pu content, and a (U, Pu) matrix coating with an intermediate Pu content, i.e. 5 ~ 10 % (Fig. 1). During normal operation, the larger Pu-rich agglomerates reach a much higher burnup than the average pellet and consequently an HBS (High Burnup Structure) restructuring can occur if the temperature of these Pu-rich agglomerates remains low enough [6]**Error! Reference source not found.**. Electron probe micro analyses highlight high concentration of fission gases in the restructured Pu-agglomerates. Fission gas (Xe, Kr) atoms are trapped within intergranular positions and formed into gas bubbles. The intergranular gas is considered to be potentially released during accident transients. This is notably the case following Reactivity-Initiated Accident (RIA) tests for which MOX fuel exhibit higher FGR compared to UO_2 for a similar burnup level [7]. Considering the two phases of an RIA transient, the behavior of fission products has a double impact:

— During the Pellet-Cladding Mechanical Interaction (PCMI) phase, the cladding remains at low temperature (< 400-500°C) and the power increase leads to pellet expansion and swelling. Here, the key parameter is the pellet swelling and so the effect of the gases during the short duration of the transient (>20s).
— The post-DNB (Departure from Nucleate Boiling) phase where the cladding temperature rises above 800°C and the rod can burst if the rod inner pressure is above the system pressure. The key parameters in this phase are the initial rod inner pressure and the transient FGR.

The main improvements to gain margins would be to decrease rod inner pressure before and during the transient to limit the risk of rod ballooning and burst. These improvements are also desired for LOCA (Loss Of Coolant Accident).

2.2. The CHROMOX Concept

The analysis of the current MOX fuel experience feedback shows that the existing margins in internal pressure in normal and accident conditions could be enhanced. This objective is of peculiar importance to keep the parity principle with UO_2 in the future. Evolutions in fuel managements, including multi recycling (see Section 4) and the progressive increase of plutonium content in MOX fuel due to the Pu isotopic vector degradation will contribute to increase FGR and rod internal pressure. Therefore, MOX evolution based on advanced microstructures is considered to provide the required performance in a complete safety approach. In 1998, a MOX fuel development programme was initiated between EDF, CEA, COGEMA and Framatome resulting in a new fuel microstructure, characterized by a near-complete homogenization of the Pu-rich phase and an increase of the average $(U, Pu)O_2$ grain size. This microstructure is obtained by doping with Cr_2O_3, giving the so-called CHROMOX microstructure (Fig.1).

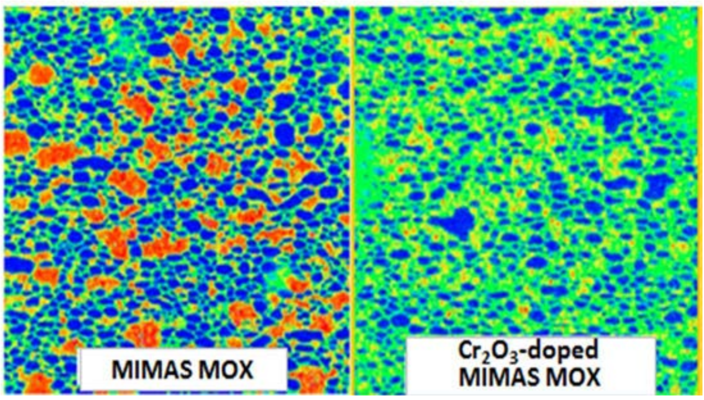

FIG. 1. Post-processes micrographs (1 mm²) of MIMAS MOX products highlighting the effect of Cr2O3-doping on the phase distribution [8] (Pu content in the different zones varies from red for primary blend agglomerates to blue for UO2 agglomerates

The use of Cr_2O_3 as a dopant in MOX fuel is drawn from the Cr_2O_3-doped UO_2 fuel pellet development programme [9]. An example of CHROMOX microstructure obtained at MELOX and characterized at CEA is shown on the right-hand side post-processed EPMA mapping of Fig. 1. It is shown that an ''inversed'' microstructure as compared to the standard MIMAS MOX product is obtained: a $(U, Pu)O_2$ solid solution containing small UO_2 agglomerates. Quantitative analyses of EPMA mappings obtained on standard and Cr_2O_3-doped MOX fuel pellets show that Cr_2O_3 doping in MOX fuel results in:

— An increase of the mean grain size in Pu-rich phases, as measured by linear intercept method, for the doped lot as compared to the standard fuel lot.
— An enhanced homogeneity characterized by smaller (U, Pu)O_2 primary blend agglomerates and increased Pu fraction in the coating phase. As a consequence of the decrease of the surface and Pu fraction in the (U, Pu)O_2 agglomerates, the coating phase is shown to be very large and containing most of the total Pu.

Additionally, the CHROMOX microstructure is expected to be beneficial regarding the extent of fuel restructuring during irradiation. The current understanding is that a MOX microstructure containing small UO_2 agglomerates for the dilution of fission products is preferred. The dilution effect is beneficial as long as the HBS formation threshold is not met. With a MOX microstructure characterized by a large coating phase, small UO_2 agglomerates and a better primary blend distribution, the CHROMOX product appears to be consistent with such a microstructural evolution.

Considering PCI, the favorable mechanical properties of MOX fuel are anticipated not to be modified because of Cr_2O_3 doping, even possibly enhanced as observed for UO_2 (due to the larger grain size) [5, 9]. Moreover, power ramp tests performed on homogeneous SBR (Short Binderless Route) MOX fuel have shown a good behavior with regard to PCI as already observed for other heterogeneous MOX fuel types [10].

The first CHROMOX-type product has been irradiated in the HALDEN test reactor up to a burnup of ~55 GWd/tHM. At end of irradiation, rod puncturing results confirm the expected benefit with respect to FGR; benefit which can be expressed as a gain of about 20 bars (hot conditions) for the MOX fuel rod internal pressure. This behavior is considered as resulting from the increased mean grain size of the fuel matrix. Furthermore, post-irradiation thermal annealing tests highlighted an improvement by ~50% (in relative) regarding transient FGR underlining the prominent role of the homogenization of the Pu distribution [8]**Error! Reference source not found.**. Following this first set of promising results and trends, the irradiation of lead fuel rods in a commercial PWR is now considered. Finally, it is to be noted that the analysis of the use of Cr_2O_3 dopant in nuclear fuel shows that it is compatible with recycling strategy.

3. MOX FUEL MANUFACTURING EVOLUTION

Orano has about 60 cumulated years of experience of (U, Pu)O_2 fuel manufacturing at industrial scale: about 30 years at the AtPu at Cadarache, with both fast neutron reactors and LWR designs based on a wide range of Pu enrichments and more than 30 years at MELOX plant for LWR designs. In both plants flexibility allowed to manufacture various designs for BWR and PWR with various Pu enrichments and pellet / cladding dimensions. During this period, about 350 tHM of MOX for LWRs have been produced at Cadarache and more than 2600 tHM at MELOX (Fig. 2). All along, Orano has gathered a unique experience of MOX plant design and operation optimization, including the specificities and constrains linked to plutonium handling. In addition, a constant effort on R&D over the years, using both internal workshops, typically the pilot line and CDA at MELOX [11] and external means in collaboration with CEA and various universities and laboratories has allowed ensuring an in-depth knowledge of the different fabrication steps and developing process parameters optimization. The use of plutonium makes MOX fuel a more complex product to manufacture compared to UO_2 fuel, requiring specific expertise. Efficient mastering of this expertise has been demonstrated by MELOX through the manufacturing of 30 million fuel pellets a year.

Regarding process optimization, since MELOX start in 1995, different evolutions of the production lines have been successfully implemented. Upon them, there was the optimization of primary blend preparation, with an improvement of pellet microstructure regarding Pu agglomerates size distribution. Another interesting step regarding fabrication is the qualification of an increase of scrap content in primary blend, which has been possible with the support of the pilot line and of the manufacturing feedback. Regarding fabrication controls, improvement and technological developments of γ-scan allow a more efficient control of fuel rod final quality.

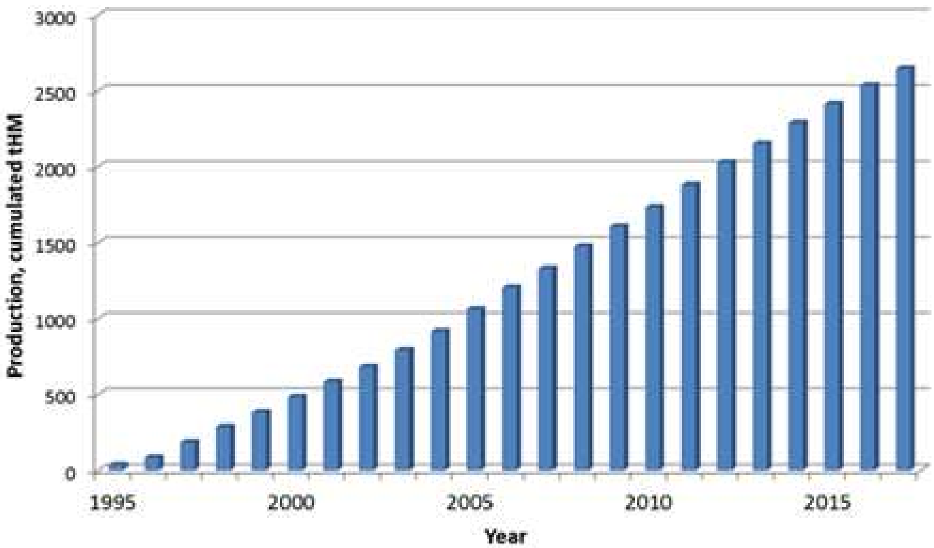

FIG. 2. Cumulated production of LWR MOX fuel at MELOX plant.

MOX manufacturing facilities require developing and implementing a safety excellence culture based on the following pillars; management of plutonium confinement and worker radiation protection. Manufacturing MOX fuel in the coming years for EDF and other customers, will lead to handle plutonium with less Pu-239 and a higher total Pu content, this being the consequences of reactor fuel management evolution (higher burnup). This means an increase in thermal and radioprotection constrains during fuel fabrication, storage and transport stages.

In order to improve management of plutonium confinement, a significant work on human factors, including glovebox and tooling optimization, training using virtual reality have been successfully developed at MELOX. Regarding radiation protection, continuous improvement has been a major driver for innovation and R&D to minimize workers exposure. MELOX has developed and implemented new materials and tools in order to increase radioprotection efficiency. Individual protections and design of the facility improvements (glove box shielding, gloves material, individual shielded glasses, etc.) together with optimization of operational and maintenance procedures allow to significantly reduce the workers exposure. In addition, new technologies such as remote handling devices and robotics are developed. As a result, in a context of constant regulation requirements evolution, these efforts have led to continuous dose exposure reduction much below the regulatory dose limitations.

Meanwhile, developing together with Framatome optimized products like CHROMOX (see Section 2) or new concepts like MIX and CORAIL-A (see Section 4) is also a driver for new improvements regarding manufacturing. For CHROMOX, first tests performed at the pilot line at MELOX allowed demonstrating the compatibility of MOX process with Cr_2O_3 doping and to assess the benefits regarding microstructure evolution. Such tests are necessary to define the optimized process parameters before fabrication at industrial scale. For CORAIL-A, adaptations may be necessary regarding radioprotection shielding and venting systems but feasibility of fabrication of such assemblies at MELOX plant has been confirmed [12]. For MIX, additional modification for criticality aspects may be necessary and industrial fabrication simultaneously with MOX is not possible.

4. PU MULTI-RECYCLING STRATEGY FOR LWRs

The reuse in LWRs of fissile materials arising from reprocessing has reached maturity and allows going further in the recycling process. Actually, there is still a great energy potential in MOX fuel after irradiation in PWRs. While working on GEN IV solutions, investigations on a shorter-term option have been launched for the reused MOX fuel to ensure a transition in current or future PWRs. The following issues need to be addressed when studying solutions of Pu multi-recycling:

- Plutonium management strategies should be consistent with maintaining high standards of safety. The requirements of nuclear safety in relation to the reactor physics properties of the various reactor systems have to be considered, which includes guaranteeing sub-criticality in all stages of the closed fuel cycle;
- Plutonium management strategies should maintain flexibility in the fuel cycle, such that future options like fast reactors are not foreclosed;
- Plutonium management strategies should be consistent with maintaining adequate security standards and safeguards arrangements to meet non-proliferation requirements, too.

4.1. Two approaches to multi-recycle Pu in PWRs

Multi-recycling strategies i.e. Pu recycling from used MOX fuel are studied based on built experience and validated codes and methods. However, current MOX assembly design needs to evolve in order to maintain the energy equivalence when degrading the Pu isotopic quality.

Since the Pu content in fuel is limited mainly for safety reasons, the solution is then to increase the number of fissions in the uranium matrix by U-235 enrichment. Two fuel assembly designs and approaches have been developed in parallel, so the best solution, technically and economically, could be defined after a thorough comparison of the respective fuel assembly performance:

- The first concept, CORAIL-A, considers the use of UO_2 and MOX fuel rods in the same assembly like for the CORAIL fuel assembly [13]. However, the original concept has been upgraded with significant modifications to improve performances regarding the use of Pu and of the reactor;
- The second, MIX, is only composed of MOX fuel rods (mono-Pu content) with enriched uranium for the mixed oxide fuel matrix instead of depleted uranium as for current MOX fuel. The enrichment of the UO_2 matrix is adaptable depending on the Pu quality and the UO_2 fuel assembly equivalence to be reached.

The analyses have been performed in 2D in an infinite medium with the code APOLLO2-A. In this study, the fuel assembly concepts are irradiated in an EPR reactor. The reference cycle selected for this study is the UO_2 equilibrium cycle — 18 months — 4.2% U-235. The average core burnup is 34 GWd/tHM and the average fuel assembly discharge burnup is 46 GWd/tHM.

The multi-recycling capacity of both fuel assembly designs has been analysed for up to five generations. One generation is considered for about 15 years and cover all the processes from fabrication, in-reactor irradiation, cooldown, transport, reprocessing, etc.

4.1.1. CORAIL-A fuel assembly concept

For the first concept, the purpose is to maximize the plutonium mass per assembly while respecting the energy equivalence to be reached. Different configurations of UO_2 and MOX rods have been tested in order to limit the peaking factors in the fuel assembly and interfaces with UO_2 fuel assemblies. The variants which were considered include the modification of the Pu content of the MOX rods depending on their location in the assembly. For example, around some guide tubes the Pu content is reduced and it is higher at the center of the assembly. However, the number of Pu contents is limited to three in order to bound fuel assembly heterogeneity. The UO_2 rods are enriched at the maximum value allowed, that is to say ~5%, therefore the spectrum is a bit harder than for current UO_2 fuel assemblies.

After testing different configurations, the configuration displayed in Fig. 3 was deemed to be the optimum with:

- UO_2 rods at the periphery to facilitate the introduction of this new fuel assembly concept in an operating reactor. The interface between the UO_2 fuel and CORAIL-A assemblies is in this way smooth, taking into account also the benefit of reduced Pu content to get a flatter power distribution;
- The Pu content is limited to 11.5%: the maximum value compatible with existing industrial manufacturing installation and operational feedback could be valued. Other limitation to account for regarding Pu content is corium sub criticality that should be ensured for all the configurations;

— Three Pu contents for the MOX rods (see Fig.3) to reduce the impact of UO_2/MOX interface within the assembly: low content (purple) on interface with UO_2 rods nearby guide tubes, medium content (light blue) and high content at the centre (yellow).

The number of UO_2 and MOX rods may be adjusted depending on the fuel cycle strategy and performance. As an example, for a 4.2% enriched uranium fuel assembly equivalence, the configuration with 141 MOX rods and 124 UO_2 rods is used.

The average Pu content and quality define the fuel assembly energy equivalence as compared to a UO_2 one. To cope with Pu degradation from one cycle to another and to have a more efficient management of the used fuel inventory, mixing of Pu contained in the various used fuels (UO_2, MOX or CORAIL-A) is considered. This mixing allows also limiting as much as possible the creation of Am-241 from decay of fissile Pu-241. After two generations the fissile Pu isotopic composition could be considered as stabilized, only the Pu-242 inventory increases slowly.

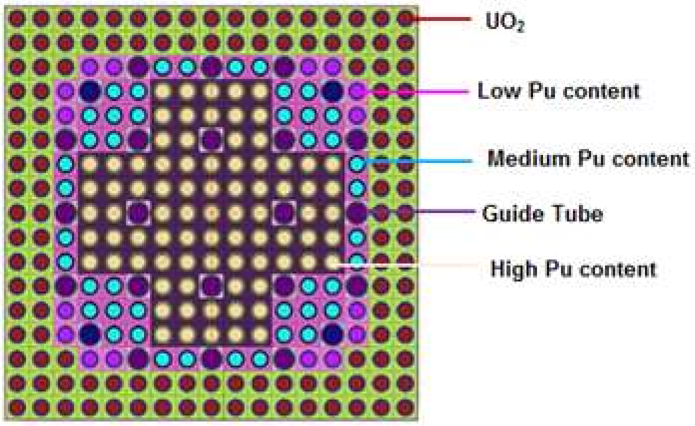

FIG. 3. CORAIL-A fuel assembly design.

4.1.2. MIX fuel assembly concept

Regarding the MIX fuel concept, Pu content and quality is determined so the economy is robust and the safety demonstration may benefit from the available feedback from MOX fuel cycles. The following parameters have been identified in order to develop a MIX fuel assembly:

— The assembly is only made of MOX fuel rods but with an enriched U-235 matrix. The uranium enrichment can be adjusted depending on the energetic and isotopic characteristics of the fuel assembly.
— The U-235 enrichment is adjusted to compensate for degradation of the Pu vector due to multi-recycling. Nonetheless, the U-235 enrichment value is limited for two particular reasons:
- The economy of the multi-recycling depends on the U-235 enrichment associated cost;
- The used MIX fuel should not contain too large amount of U-235 so current reprocessed uranium recycling supply chain could be used (about 1.2 % max).

— The Pu content is limited to overcome any safety issues, mainly related to void coefficient and corium criticality.
— In order to be able to use as much Pu as possible per fuel assembly, therefore reducing the reactor fleet mobilized for multi-recycling, the Pu quality is controlled by mixing Pu from used MIX or MOX fuel and used UO_2 fuel.
— After two generations the Pu isotopic composition could be considered as stabilized, meaning that the U-235 enrichment of the UO_2 matrix is also constant; only the Pu-242 inventory increases slowly.

As an example of MIX assembly, fuel assembly studies were done on the basis of an equivalent 4.2% enriched UO_2 fuel assembly for EPR. The characteristics obtained for this MIX fuel assembly are the following: 9.5% Pu content for all the rods - 2% U-235 enrichment - 50% Pu from used MOX or MIX and 50% used UO_2.

4.1.3. Comparison of the reactor performance of the two fuel designs

Main reactor parameters for both concepts were compared:

— Void coefficient: Both concepts respect this important safety criterion. A sensitivity study to the boron concentration was performed (2000 to 3000 ppm). For the MIX assembly the improvement of the Pu quality thanks to the use of Pu from used UO_2 fuel allows to upload rods with 9.5% Pu. Otherwise the Pu content should have been reduced and uranium enrichment increased for the sake of offset. Regarding the CORAIL-A concept, no difficulty is encountered because of the use of UO_2 rods;
— RCCA efficiency for both concepts is comparable but slightly better for the CORAIL-A concept which again benefits of the presence of UO_2 rods;
— The interface between UO_2 and MOX rods may create high peaking factors especially in case of the CORAIL-A design. The proposed Pu zoning allows to overcome that and to fulfil the design criterion imposed (peak factor ≤ 1.15). For the MIX fuel assembly there is not such an interface for equilibrium fuel management. However, the fuel assembly zoning may be required for the transition fuel management;
— Moderator temperature coefficient has been determined using current validated methodology. In all configurations and for both designs, the results show that the moderator temperature coefficient is always negative. Sensitivity analysis to the boron concentration with enveloped values confirmed the results.

The studies reported above highlight that for both concepts high performance regarding fuel cycle length or burnup could be reached by controlling the Pu content and/or the isotopic composition depending on the defined cycle strategy. The feasibility for multi-recycling up to 4–5 fuel generations of about 15 years each has been demonstrated. The stabilization of the Pu isotopic composition is reached after about 2 generations when mixing CORAIL-A or MIX used fuel and UOX used fuel. The analyses performed are based on an EPR reactor and 100% MIX or CORAIL-A fuel management. However, the results obtained and our significant experience with 30% MOX fuel management emphasize the possibility to proceed also with a 30% multi-recycling CORAIL-A or MIX fuel management. The fuel assembly choice will depend on the available reactor fleet and its performances, the manufacturing capabilities and the used fuel management strategy.

Both concepts contribute to reduce the available Pu stock because they burn Pu during the cycle. The downloaded U enrichment is lower than the limit accepted by the fuel cycle facilities, about 1% max. As an example here below the figures for consumed Pu based on fuel assembly calculations for an EPR standard fuel cycle:

— CORAIL-A: about 18 kg of Pu per TWh(e) for all generations;
— MIX: about 48 kg of Pu per TWh(e) for all generations.

5. CONCLUSION

In more than 60 years of Framatome and Orano experience, Pu recycling through fabrication and irradiation of MOX fuel in LWRs has demonstrated the reliability of this process allowing managing at best uranium resources. The operational performance of MOX fuel rods and assemblies has been assessed by numerous pool-site inspection and hot-cell measurement campaigns. This unique feedback experience points out that MOX fuel behaviour is as good as UO_2 fuel. However, the high linear heat generation rates seen by the MOX fuel rods induce a somewhat higher fission gas release and subsequent reduction of rod internal pressure margins to safety criterion. To maintain in a sustain way the energy parity principle between MOX and UO_2 fuel, product development is considered based on advanced microstructures. In that respect, the CHROMOX product which involves Cr_2O_3 doping is characterized by an enhanced homogeneity of the Pu distribution in the fuel and an increased matrix grain size. With these evolutions, larger internal pressure margins are anticipated as well as some enhancement in the retention of gaseous fission products in accidental conditions by reduction of restructured areas.

The design and operation of the MELOX MOX production plant takes account for the most recent and stringent safety requirements regarding confinement of high radioactive materials, protection of workers and environment from exposure, safety criticality, etc. To support the MOX fuel development in the future and maintain energetic equivalence with UO_2 fuel in more demanding core managements, some adaptations are already considered. The objectives are to be able to manufacture the CHROMOX product and also to face the inherent degradation of the Pu isotopic vector of MOX fuel and its higher Pu content from increased core management cycle length.

On this last point, Pu multi-recycling strategies in LWRs are studied with new fuel technologies. In order to be able to use low quality Pu in a PWR spectrum, fissile uranium needs to be added. With the CORAIL-A option, developed by Framatome and Orano, the assembly contains about half of MOX fuel rods and the remaining as UO_2 rods. By contrast, the MIX fuel assembly contains only MOX rods with an enriched uranium matrix that compensates the Pu degradation. Development of those fuel technologies, that could be coupled with the most advanced Framatome fuel assembly design GAIA for PWRs, will offer flexibility to switch to future technically and economically robust advanced cycles in current or future LWRs with a limited impact to the reactor design and its performance. Thus, these developments will allow implementing efficient solutions bridging the gap with the potential development of Gen IV reactors. Any solution of advanced fuel developed within the closed cycle strategy integrates aspects of the whole fuel cycle, including the recycling of such fuel.

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3.5. TRACK 5 – IMPACT OF ADVANCED NUCLEAR ENERGY SYSTEMS ON THE BACK END OF THE FUEL CYCLE

Overview prepared by A. Barry (Canada) and A. Bychkov (Russian Federation), **Track Leaders**

This track considers the complex aspects of advanced reactor system development in terms of reducing the impact of nuclear power on the environment.

In the first session, five presentations described the results of multilateral studies and national programmes in the field of advanced fuel cycles. All presentations were related to advanced nuclear reactors and SF management approaches to reduce the amount of radioactive waste, decay heat, and radiotoxicity. It is shown that the partitioning and recycling/transmutation of plutonium and minor actinides reduces these parameters.

The report "Back End Fuel Cycle Strategies in uncertain Generation-IV futures" summarizes some multilateral studies on the impact of a combination of different reactor systems to achieve a harmonious nuclear closed fuel cycle, in particular, the main findings of the SYNERGIES collaborative project completed in the framework of INPRO.

The European R&D project GENIORS focused on the development of an efficient system for plutonium multi-recycling and minor actinides recycling as MOX fuel in Gen-IV reactors was described in detail, including the importance of Technology Readiness Level (TRL) of both reactor and associated fuel cycle.

The national R&D programmes of Japan, Russian Federation and India highlighted the development of advanced fuel cycle with new reactor systems (fast reactor with sodium or lead coolant, Accelerator-Driven Systems (ADS)) and new approaches to actinide recycling (in Molten Salt Reactors (MSR)). The specific possibilities of industrial and medical use of fission products contained in spent fuel were noted. The reported results show a promising opportunity to apply advanced technologies to create effective nuclear fuel cycles with a minimum amount of radioactive waste.

The second session included reports that highlighted several solutions in this area: the development of partitioning of high level waste (HLW), trends in the development of molten salt reactors and ADS. The R&D results of studies on minimization of the volume of HLW disposal sites by application of actinide recycle were reported. The presentation on some implications of accident-tolerant fuel (ATF) technology on spent fuel also aroused great interest and the need to embark spent fuel management considerations, including storage but also impact on reprocessing or disposal from the ATF design phase shared by participants.

Session 5.1: Impacts of advanced nuclear energy systems on the back end of the fuel cycle

Session Chairs: A. Bychkov (Russian Federation) and T. Okamura (Japan)

Session 5.1 comprised of five presentations, one from Belgium, one from Japan, one from France, one from Russian Federation and one from India.

- **Paper ID#139 by L. Van Den Durpel (Belgium)** presented the overview of strategic studies in the frame of different international projects as IAEA INPRO Project and EU projects. Various authors studied SFM options described as prospects for post-2050. The expectations were connected with "Gen-IV" systems or even more advanced "Generation-X" (partitioning & transmutation (P&T)). The authors are considering which back end fuel cycle strategies may construe a proper solution-oriented SFM aligned to nuclear energy's role within the uncertain prospect to evolve soon towards advanced systems.
- **Paper ID#167 by K. Tsujimoto (Japan)** presented the national studies on Partitioning and Transmutation technology that is expected to be effective to mitigate the burden of HLW disposal by reducing the radiological toxicity and heat generation. This R&D activity is based on the Strategic Energy Plan of Japan. The Japan Atomic Energy Agency (JAEA) continues to work on two concepts: homogeneous recycling of minor actinide in fast reactors and "double-strata" strategy, using an accelerator-driven system. Different processes were considered for partitioning of minor actinides from spent fuel and uranium-free nitride fuel was chosen as the first candidate for ADS.
- **Paper ID#28 by S. Bourg (France)** presented the content and current results of European R&D Project GENIORS. The project addresses research and innovation in fuel cycle chemistry and physics for the optimization of MOX fuel potentially containing minor actinides in multi-recycling strategies in GEN IV reactors. By implementing a three-step approach (reinforcement of the scientific knowledge, process development and testing, system studies, safety and integration), GENIORS will lead to the provision of more science-based strategies for nuclear fuel management in the European Union.
- **Paper ID#68 *(Invited)* by A. Shadrin (Russian Federation)** presented Russian R&D activity on closed nuclear fuel cycle with fast reactors. The current status "PRORYV" project described. It includes the construction of pilot demonstration power complex with reactor BREST-OD-300 with lead coolant and closed fuel cycle facilities: for fuel fabrication/refabrication, for mixed uranium-plutonium nitride spent nuclear fuel reprocessing and radwaste management. The results of experiments and models and codes development were presented.
- **Paper ID#193 by C.P. Kaushik (India)** presented the Indian R&D activity in advanced fuel cycle with reprocessing and high level radioactive treatment. Additional to recycling of the basic components of spent fuel (uranium and plutonium), the extraction and application of some fission products (Cs-137, Sr-90, Y-90, Ru-106) were considered and demonstrated. This complex approach could reduce HLW volume.

Session 5.2: Impacts of advanced nuclear energy systems on the back end of the fuel cycle

Session Chairs: S. Bourg (France) and A. Barry (Canada)

Session 5.2 comprised of five presentations, one from United Kingdom, two from Japan, one from China, and one from Russian Federation.

- **Paper ID#62** ***(Invited)*** **by D. Goddard (United Kingdom)** presented an overview of various accident tolerant fuels technologies and their implications on spent fuel management. ATF technologies were categorized with new claddings and new fuel types. Most ATF concepts should perform well both in storage and in geological disposal as current fuel designs, but confirmatory test is required.
- **Paper ID#170 by T. Sugawara (Japan)** presented an accelerator-driven system for partitioning technology, specifically building on previous work on the ideal fuel composition to realistically include U that will accompany Pu in the partitioning process and MA in the partitioning and reprocessing process. The k-eff value, the proton beam, and relative lattice parameter difference (RLPD) were found to be appropriate.
- **Paper ID#172** ***(Invited)*** **by C. Xu (China)** presented work on partitioning of high level liquid waste in China. Successful hot tests of the trialkyl phosphine oxides (TRPO) process led to the construction of a pilot test facility. Additional hot tests were complete to make improvements to the process to address problems in Cs removal.
- **Paper ID#8 by V. Ignatiev (Russian Federation)** presented a molten salt reactor to manage minor actinides from VVER 1000/1200 fuel. Significant work has been completed on critical viability issues such as material compatibility, salt physical and chemical properties, fuel salt processing capabilities, and intrinsic safety. Future work requires experimental infrastructure, including a proposed test reactor prior to a large-scale power plant.
- **Paper ID#34** ***(Young Generation Challenge Winner)*** **by T. Okamura (Japan)** presented the reduction in geological disposal area that can be achieved via partitioning technologies. Results showed that fuel burnup had a minimal effect on the amount of vitrified waste, but the contribution to heat generation in vitrified waste from high burnup fuel increased for short-lived Cs-137, Sr-90, and their daughter nuclides, but decreased for Am-241. Separation of Cs and Sr resulted in significant reduction on the deep geological repository (DGR) footprint.

Paper ID#68
FAST REACTOR SNF REPROCESSING FOR CLOSED NUCLEAR FUEL CYCLE

A. Y. SHADRIN, K.N. DVOEGLAZOV, V.L. VIDANOV, O.V., YU.S. MOCHALOV
Innovation and Technology Center of the Project "PRORYV"
Moscow, Russian Federation
Email: shau@proryv2020.ru

V.I. VOLK, V.A. KASHCHEEV, O.A. USTINOV
BOCHVAR INSTITUTE
Moscow, Russian Federation

ABSTRACT

Closed nuclear fuel cycle (CNFC) with inherent safety fast reactors (FR) is a new integrated product in the branch of atomic energy. A pilot demonstration power complex with reactor unit (BREST-OD-300) with lead coolant is under construction at the Siberian Chemical Combine in frame of the "PRORYV" project. This pilot demonstration power complex includes not only fast reactor but also facility for fuel fabrication/ refabrication and facility for mixed uranium-plutonium nitride (MNIT) spent nuclear fuel (SNF) reprocessing and radwaste management. Integrated system of models and codes for the coordinated simulation of different processes and phenomena for CNFC technologies are also under development.

1. INTRODUCTION

Closed Nuclear Fuel Cycle (CNFC) with Fast Reactors (FRs) is a new integrated product in atomic energy. CNFC allows:

- to reduce an amount of accumulated used nuclear fuel;
- to manage a radioactive waste on the principles of radiation equivalency;
- to realized technological support for the non-proliferation;
- to provide the cost efficiency and compare with other energy sources.

Industrial scale CNFC should be realized at the experimental demonstration power complex (EDEC) including reactor unit BREST-OD-300, mixed uranium and plutonium fuel fabrication unit and FR SNF reprocessing unit. EDEC is and development and constriction under project "PRORYV".

Two versions (combined (pyro + hydro) and hydrometallurgical) of FR mixed uranium-plutonium nitride and oxide SNF reprocessing technology are under developing within "Proryv" project. Up to day the R&D programme on hydrometallurgical technology is close to complete. The study of pyroelectrochemical and plasma separation technologies are at the different R&D stages.

The following results were achieved for FR SNF hydrometallurgical reprocessing:

- a pilot set-up of extraction and crystallization affinage of U+Pu+Np mixture with the full-scale crystallizer was created;
- main technological equipment for hydrometallurgical reprocessing of MNIT and MOX SNF FR were developed;
- a realized test of processes and equipment for extraction and crystallization affinage of U+Pu+Np mixture confirmed a total decontamination factor 5×106;
- deep recovery (> 99.9%) of actinides was demonstrated;
- full-scale set-up for microwave denitration of U+Pu+Np, U-Am, U-Cm were developed and tested;
- partitioning technology for group separation of rare earth elements and transplutonium elements and for Am/Cm separation were tested using irradiated fuel;
- full-scale prototypes of industrial equipment (dissolution, clarification, off-gas cleaning, crystallization, microwave denitration, Am/Cm separation) were developed and tested;

— dry separation of SNF and fuel cladding, removal of more than 99.9% tritium and more than 98% C-14 were demonstrated for the MNIT SNF voloxidation process;
— off-gas cleaning technologies provided recovery > 99.99% I, 99% H-3 and C-14, were tested.

In terms of radwaste management for hydrometallurgy and the combine technologies for MNIT and MOX SNF were developed and verified including cold crucible for vitrification HLW and combination of alkali precipitation and tangential filtration for decontamination of U-Pu-Am-containing solutions.

The integrated system of models and codes is under development for simulation of heterogeneous processes and phenomena that are required to consider under calculating maintaining and reasoning the safety of CNFC technologies.

2. COMBINED (HYRO + HYDRO) TECHNOLOGY FOR FR SNF REPROCESSING

2.1. Head-end and pyrochemical operations

The EDEC aims include the demonstration and development of CNFC technologies with lead cooled reactor and MNIT fuel. The unit for MNIT fuel fabrication/ refabrication is under construction now. The technologies of MNIT SNF reprocessing is under development now. The combine (pyro + hydro) technology (PH-process) is chosen for as a basic for EDEC. PH-process includes high temperature head-end operations, pyrochemical recovery of actinides and hydrometallurgical operations including deep purification of recycled actinides and recovery and separation americium and curium (Fig. 1). The most detailed description is given in [1]. Progress and results of 2017-2018 are given in [2].

The voloxidation of MNIT is a first chemical operation after decladding and mechanical fragmentation. This operation allows to remove H-3, iodine and C-14 almost totally. The experiments on oxidation used MNIT were made in RIAR in 2017–2018. It was shown that MNIT can be fully oxidized and removal of 99.8% and more of H-3, and 98.4% and more of C-14 were achieved (Fig. 2).

The description of off-gas decontamination operation is given in [3, 4]. The recovery of 98.2–99.9% C-14 is necessary for environment safety [5]. The developed technology allows to achieve the necessary level. The recovery of 99.99% all iodine forms can be achieved. The optimal sorption temperature is 160–170°C.

The demonstration of pyrochemical operations with real (irradiated) MNIT was performed in air hot cells in 2013. However, the passivation of MNIT SNF pellets due to formation of low soluble in electrolyte UNCl surface layer, and, probably, UN/U_2N_3 formation also insoluble in electrolyte, could lead to losses of uranium. Due to these and some other reasons these experiments were not fully successful. The oxidation MNIT to actinides oxide powder and reduction of oxides to metal actinides was proposed. The reason of this approach was experimentally confirmed with simulated products, U and Pu, and with irradiated SNF.

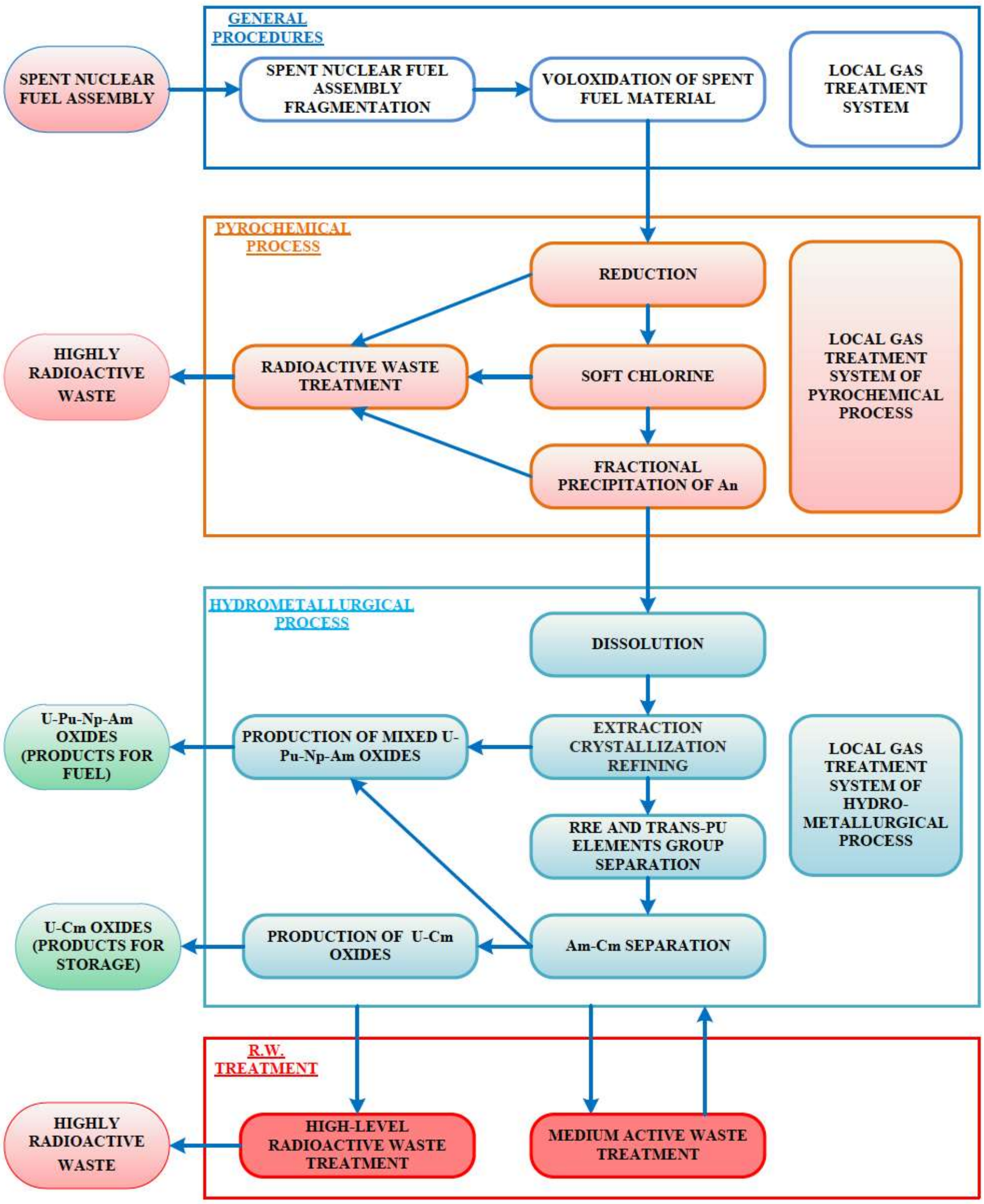

FIG. 1 PH-process principal flow-sheet.

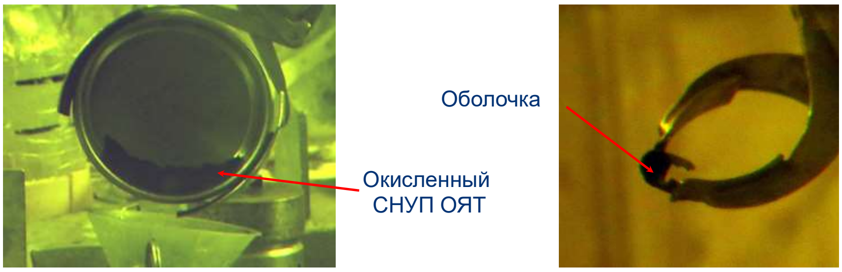

FIG 2. Oxidized MNIT SNF.

2.2. Hydrometallurgical operations

The recovery and purification of U-Pu-Np product by extraction and crystallisation was checked at Siberian Chemical Combine set-up. This set-up contains from 6 blocks of centrifugal contactors, two separators and crystallizer. The recovery of U, Pu and Np was more than 99.97%. The Pu decontamination factors for Cs (106), Sr (103), Zr and Mo (103), and rare earth elements (104) were achieved. The crystallisation gives an additional decontamination factor 100–200. The direct denitration of U and Pu nitrates allows to prepare the needed products without additional amount of secondary waste. Mixed oxides of U and Pu were prepared using laboratory set-up for microwave denitration for MNIT fuel fabrication. Apart from mixed U and Ce oxides (several kilogram) were prepared using experimental set-up for microwave denitration (Fig. 3). U-Ce oxides (Fig. 4) were used for mixed nitride synthesis and for MNIT pellets fabrication. The prepared pellets are satisfied to technical requirements for MNIT fuel.

FIG. 3. Experimental set-up for microwave denitration.

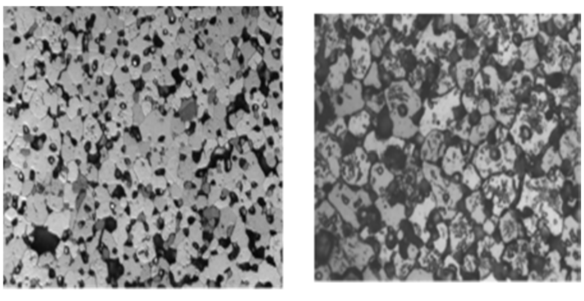

FIG. 4. Microstructure of UN and U(Ce)N pellets.

Results of studies on partitioning performed at 2016–2018 are given below. 99.99% of Am were recovered from high level waste (HLW) simulated solution by dynamic test using solution of TODGA in metanitrobenzotrifluoride. 99.9% of Am and Cm were recovered from real HLW using some extraction system. The technology of americium and curium separation was demonstrated in 2016 at the trial and industrial unit of PO “MAYAK”. Around 14 g of Cm-244 were recovered of which 9 g was the fraction of enriched curium with the americium content of less than 6% by activity. The mixed americium-curium fraction contained around 4.6 g of Cm-244 and around 40g of Am-241 and Am-243. The enriched americium fraction the curium share was less than 0.8% by mass, and the content of Eu-154 and Eu-155 was less than 0.1% by activity.

3. MANAGEMENT WITH RADIOACTIVE WASTE

The full-scale prototype of industrial “cold” crucible for HLW vitrification was made and tested in Bochvar Institute (Fig. 5). This set-up with “cold” crucible not only has a remote control but this crucible can be distantly removed and changed for a new one. The test was performed for 300 hours including 200 hours of continuous operation. No deviations from technical requirements were obtained.

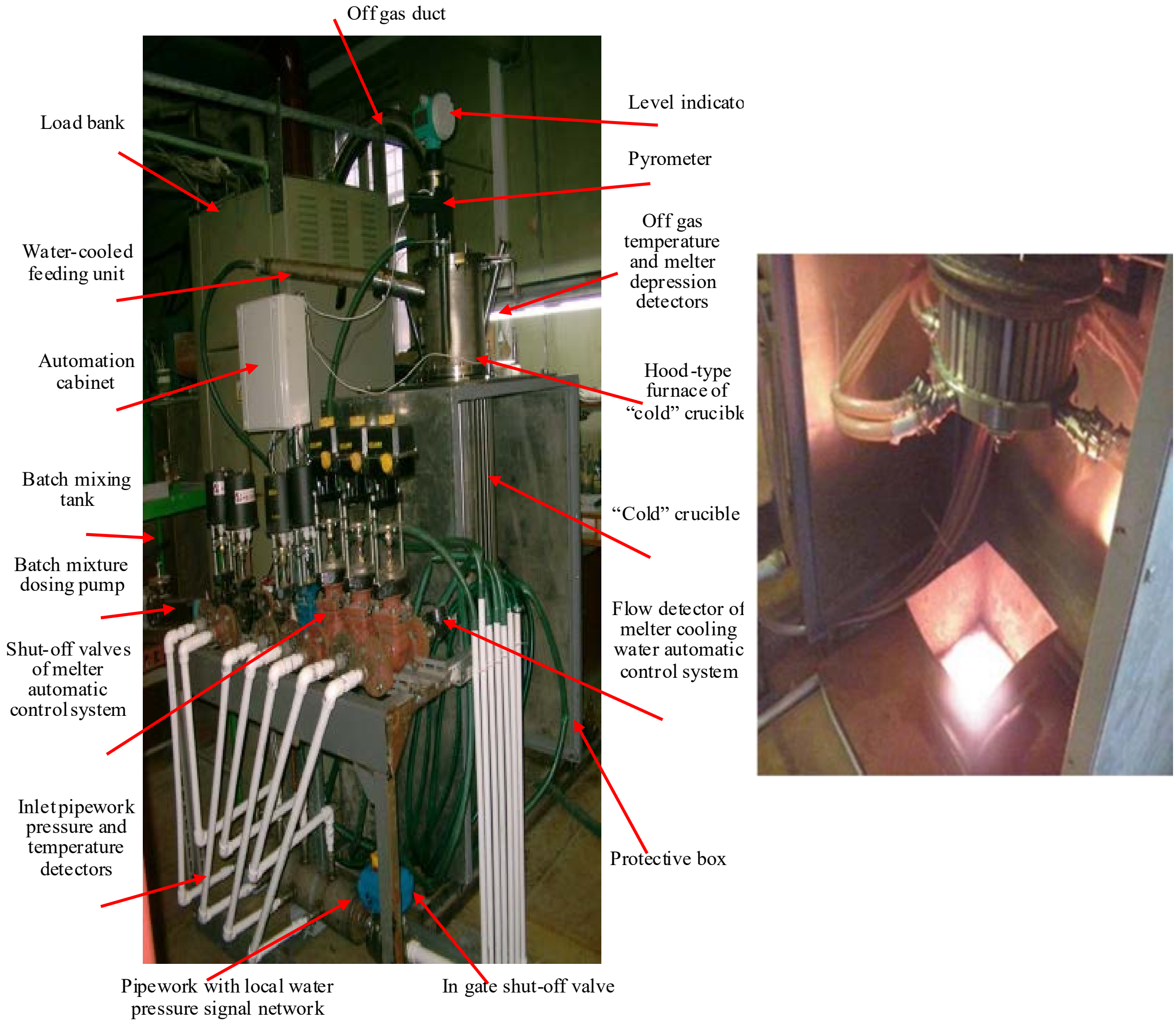

FIG. 5. Full-scale prototype of set-up with “cold” crucible for HLW vitrification (a) and the drain of glass (b).

4. INTEGRATED SYSTEM OF CNFC MODELS AND CODES

The development of SNF reprocessing technologies is accompanied by the development of a system of models and codes for the mathematical simulation of technologies, namely:

- Code VIZART for the calculation of the material flows of the technological scheme and its sections in the stationary and dynamic modes, taking into account the evolution of the isotopic composition;
- Mathematical models of technological processes;
- Code COD TP for simulation of the operation of technological schemes in real time (model of automated process control system), including modeling of emergency situations;
- Simulation (kinematic) model of technological operations.

The state of development of the system of models and codes is described in [6].

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Paper ID#62
SPENT FUEL MANAGEMENT CONSIDERATIONS FOR ACCIDENT TOLERANT FUELS

J. LUCAS, D.T. GODDARD, D.I. HAMBLEY
National Nuclear Laboratory, Springfields, United Kingdom
Email: joel.lucas@nnl.co.uk

Abstract

The accident at the Fukushima-Daiichi plant in Japan in 2011 highlighted vulnerabilities in the current zirconium (Zr) alloy clad uranium dioxide (UO_2) fuel to an extended loss of cooling. Improving the resilience of the fuel and cladding is considered a high priority for the nuclear industry and has resulted in significant research into the development of so-called Accident Tolerant Fuels (ATF). ATF are widely expected to be deployed in the near future in existing and future Light Water Reactors (LWRs). Post discharge management and dispositioning of spent ATF is a topic that must be addressed in order to demonstrate responsible management of the fuel cycle and yet has received little attention to date. In this review the spent fuel management considerations of several leading ATF fuel and cladding concepts are assessed against current LWR fuels. The concepts include coated Zr alloys, advanced iron alloys and silicon carbide composite claddings and advanced UO_2 and high uranium density fuels. Technical challenges regarding each different material are highlighted; particularly focusing on reactivity and durability in water. Recommendations are made where variations of current storage procedures are likely to be required.

1. INTRODUCTION

Accident Tolerant Fuels (ATF) are being developed to improve the resilience of LWR fuels under accident conditions. This is being achieved largely through the development of more oxidation resistant cladding materials, although improvements in the thermal conductivity of the fuel can also reduce pellet centreline temperatures and increase the power-to-melt. Such considerations have also led to interest in higher density fuel materials, which despite their higher reactivity to water compared to UO_2, compensate for the increased costs associated with the more robust claddings materials. Whilst the performance of ATF during reactor operations is of primary importance during early stage development, as these concepts mature it becomes important to understand potential impacts on the whole fuel cycle to avoid creating unexpected back end costs and/or leaving the industry exposed to public challenges on issues of responsible management and sustainability [1]. This paper provides an initial overview of the likely performance of ATF during post-irradiation storage and disposal.

The candidate concepts for ATF cladding include a near term option of coating the existing Zr alloy cladding as well as longer term options of Fe based alloys, (drawing on past experience of austenitic stainless steels) and ceramic silicon carbide composites. Candidate fuels include advanced (higher thermal conductivity) UO_2 and higher density fuels including uranium nitride (UN) and uranium silicide (U_3Si_2)

After discharge from reactor, all spent LWR fuel is stored for an initial cooling period in station ponds. The pond water is typically kept at a temperature of below 50°C in order to maintain adequate cooling of the fuel and minimise evaporation. The pond water chemistry is closely controlled to minimise corrosion and is constantly monitored.

There are two principle options for management of spent fuel: disposal or reprocessing. Both options involve interim storage and transport. Fuel can be shipped from station cooling ponds once the heat generation rate has fallen to an acceptable level for transport. This can be as short as a few months to a few years, depending on the fuel type, irradiation and the transport cask design.

LWR fuel can be reprocessed a few years after discharge from reactor. Fuel is typically transferred to storage ponds at reprocessing plants prior to reprocessing as this allows the flow of materials to be buffered and allows optimisation of the materials entering reprocessing. Reprocessing of fuels is not considered in detail in this paper as it entails dismemberment and dissolution of the fuel assemblies. Disposal is, however, considered because it involves emplacement of the fuel assemblies into a disposal facility and therefore the characteristics of the fuel cladding and fuel matrix have an impact on subsequent behaviour and activity release.

Compared with reprocessing, spent fuel needs to be stored for much longer periods of time before the heat generation rate falls to a level considered acceptable for disposal. The minimum cooling period varies with fuel irradiation and disposal concept but is typically many decades. The quantity of fuel being stored, and the duration of storage are therefore greater for this option. Long term storage of spent nuclear fuel can be conducted in either in ponds (wet storage) or in dry stores and a range of storage options are available for this [1]. Typically, wet storage is undertaken in similar conditions to those used at power stations. Dry storage typically involves drying of the fuel and sealing into a container with an inert gas. This leads to higher fuel temperatures during the drying process and during the initial dry storage period, than are experienced in pond storage. Most dry storage systems employ passive cooling and therefore require less infrastructure than pond storage to maintain cooling and hence containment. There are a wide range of potential degradation mechanisms that can affect fuel storage systems and a number of reviews of fuel degradation mechanisms have been conducted in the past 10 years (e.g. [3–5]). For current Zr alloy clad UO_2 fuel the following degradation mechanisms are considered most important in wet storage:

— Uniform (aqueous) corrosion;
— Localized corrosion (Pitting, galvanic, microbial induced corrosion (MIC));
— Hydriding;

whereas the dominant degradation mechanisms in dry storage are [1]:

— Air oxidation;
— Thermal creep;
— Stress corrosion cracking (SCC);
— Delayed hydride cracking (DHC);
— Hydride reorientation; and
— Hydrogen migration and redistribution.

Although operating conditions in LWRs are more severe than in storage the potential exposure times are much longer (and more so in disposal). Testing of newly developed ATF at low temperatures relevant to storage and disposal conditions will be important for evaluating the impact on back end fuel cycle options.

In geological disposal concepts, fuel is dried and encased in high integrity canisters that provide containment for the fuel and prevent contact between the fuel and groundwater for several thousand years. The disposal container designs associated with well-developed disposal systems are, in general, smaller than those most commonly being loaded with LWR fuel for dry storage.

2. CLADDING

2.1. Coated Zr Alloys

The addition of a protective coating on the outer surface of the Zr-alloy cladding is a relatively simple but effective method of reducing the rate of oxidation of the Zr alloy in high temperature steam that would be encountered during a Loss of Coolant Accident (LOCA). Whilst these coatings are not expected to be able to protect the fuel indefinitely, they are seen as a means of increasing the coping time during which operators may be able to restore cooling and save the reactor from core melt. A wide range of coating materials have been examined, particularly those which form protective chromia, alumina or silica layers when exposed to high temperature steam. Although alumina provides the best high temperature oxidation resistance, it is less stable in reactor operating conditions and therefore most focus is currently on chromium [6]. Cr coated Zr alloys have demonstrated an approximately 10-fold reduction in oxidation due to the production of a Cr_2O_3 protective layer, with commercial irradiation of test rods already in progress [7]. Reduced corrosion also leads to less hydrogen production compared to standard cladding. This has potentially beneficial consequences because hydrogen generated by fuel corrosion during irradiation leads to the formation of zirconium hydride platelets in the cladding. During drying prior to dry storage, the temperature can increase to between 250°C and 400°C. When temperatures are at the higher end of this range, hydrogen dissolution can occur as the solubility of the hydrogen is increased. When cooled, the hydrides will re-precipitate and favour a radial orientation due to the internal pressure of the fuel pins caused by azimuthal stresses. These radial hydrides significantly increase the probability of stress related failure of the fuel pins. In order to mitigate this, the amount of hydrogen allowed in the fabricated Zr alloy is very low (~1.5 ppm) [8] and limits are applied to the maximum cladding temperature during drying and the number of drying cycles that can be used.

Corrosion during pond storage is also expected to be decreased when compared to uncoated Zr alloys. Eventual dissolution of the cladding could lead to localised loss of Cr which could subsequently lead to localised corrosion of the Zr alloy. For a disposal system this is unlikely to be of concern as current performance assessments do not claim credit for cladding integrity.

Dissolution of Cr can lead to the formation of CrO_4^{2-} which is a strongly oxidising species, which could affect the environmental conditions experienced by the fuel matrix if it were to accumulate within a repository. In water CrO_4^{2-} formation is thermodynamically stable at electrode potentials of around 100 mV(SHE) at pH 12 to around 600 mV(SHE) at pH 7 [9] in the absence of hydrogen peroxide. Therefore, its formation in repository is unlikely [10], but cannot be precluded without an understanding of expected groundwater chemistry and radiolysis effects.

2.2. Advanced Fe Alloys

Austenitic stainless steels were used as cladding materials in LWRs until the 1980's prior to the widespread adoption of Zr alloys. The preference for Zr alloys is largely an economic one, due to its low neutron capture cross-section. Estimates suggest a cost penalty of 15–35% for steel cladding compared to Zr alloys even with a reduction in cladding thickness to around 300µm [11]. However, Fe based alloys containing Al exhibit excellent oxidation resistance due to the formation of an alumina protective layer, up to temperatures close to the melting point of the alloy (1475°C). Incidentally, under normal operating conditions chromia layers are formed preferentially, providing adequate protection. There are commercial alloys such as Kanthal APMT with the nominal composition of Fe-21wt.%Cr-5wt%Al-3wt%Mo, but other alloy compositions are under development [13].

Spent fuel storage experience of steels (types 304, 304L, 34 and 348H) from five US LWRs, totalling >2000 fuel assemblies was summarised in an EPRI report in 1996 [14]. No unexpected behaviour was noted for these spent fuel assemblies during pond storage over a period of up to 25 years. Uniform corrosion rates during wet storage (<50°C) were predicted to result in a reduction in wall thickness of 15µm over a 50-year period. The behaviour of these materials, and claddings of similar composition, is expected to be similar to other stainless steels (such as those used for UK Advanced Gas cooled Reactor (AGR) fuel cladding). Experience of storing AGR fuel in the UK indicates no discernible loss of wall thickness after several decades of storage in the presence of corrosion inhibitor which provides further confidence in the long term wet storage of stainless-steel clad fuels.

Compared to Zr alloys, iron alloys do not readily absorb hydrogen so embrittlement is not a significant issue. In general, steel cladding has a higher acceptable temperature during drying than Zr alloy cladding as it is not affected by hydride reorientation.

However, iron alloys, especially those with higher Cr content, are susceptible to sensitisation mechanisms (both thermal and irradiation assisted). Thermal sensitisation characterised by the formation of chromium carbide precipitates and depletion of Cr in solution is not expected to affect LWR fuels since temperatures in excess of 427°C (800°F) are required and the alloy can be manufactured to be largely resistance to thermal sensitisation. Neutron irradiation leads to Radiation Induced Segregation (RIS) as a result of a dynamic equilibrium in atomic migration of vacancies and atoms at grain boundaries. Under a limited range of temperatures, including those encountered by AGR fuel, this can lead to significant depletion of Cr at grain boundaries [15–17], leaving the material susceptible to localised corrosion, particularly stress corrosion cracking (SCC) [18]. However, there appears to be mechanistic differences between SCC in gaseous environments and in water that result in more severe and extensive corrosion is low humidity 'dry' storage conditions. Currently dry storage of AGR fuel is not underpinned and further work would be required to underpin dry storage of other RIS affected fuel claddings in dry conditions in order to avoid failures.

Unlike Zr alloys, which are effective hydrogen getters, Fe based alloys will allow hydrogen (and similarly tritium) to permeate. The consequence of this both in reactor as well as for long term storage will need to be evaluated. A liner material could potentially be used as a barrier to tritium release.

2.3. Silicon Carbide Composites (SiC/SiC)

SiC/SiC composites consist of high purity SiC fibres wound and braided into tubular forms and then impregnated with a matrix of SiC. The oxidation resistance of these materials in high temperature steam is superior to both the coated Zr alloys and Fe based alloy ATF options. However, the lack of any prior experience in LWRs and the fact that the material behaves more like a ceramic than a metallic cladding, means that its introduction as an ATF concept is particularly challenging. One area of concern is hydrothermal corrosion under normal operating conditions due to the difficulty in forming a stable oxide layer. This is particularly true of the oxidising conditions present in Boiling Water Reactor (BWR) water chemistries. One potential solution is to dose with hydrogen as is typical for Pressurised Water Reactors (PWRs), however if this is not sufficiently effective an additional protective coating may be required.

SiC is very corrosion resistant at low temperatures in water and air as a result of the formation of a protective SiO_2 surface layer, although irradiation damage has been found increase the rate of dissolution. Experimental work on TRISO particles has been carried out (both irradiated and unirradiated) which suggested that the SiC layer has long term durability in repository conditions [19,20].

Since SiC/SiC cladding contains a source of carbon, there is potential for the formation of organic molecules or gaseous species. SiC/SiC cladding may therefore add to the source term for carbon in a repository and the form of carbon and its rate of generation will be of interest for the performance assessment. Whilst it is not currently possible to assume that the SiC/SiC cladding is fully corrosion resistant under repository conditions it is clear that it has a sufficiently high durability that any additional source of carbon will not form an acute release and there is no reason to consider that it would be unacceptable within the current Geological Disposal Facility (GDF) concepts. The evolution of C from the fuel cladding would nevertheless require work to identify the potential corrosion products and the rate of production under relevant disposal conditions at an appropriate point prior to a disposability assessment.

A UC_2 kernel fuel coated with a layer of pyrolytic carbon encased in loosely sintered SiC within a SiC cladding tube were tested in the UK Windscale Advanced Gas-cooled Reactor (WAGR) on an experimental basis in the late 1960s. During testing failure in the cladding was detected at low burn up. The pins were placed straight into pond storage and have recently been migrated to dry storage. No further signs of corrosion were identified during repackaging. A few fuel pins did fracture but these are thought to have been caused during the packing process rather than the irradiation or storage.

Although this WAGR testing provides a good start point for exploring the suitability of SiC/SiC in storage, it was carried out on a small scale for monolithic SiC materials, so testing of modern composite materials irradiated to levels commensurate with modern fuel is recommended to provide confidence in the effects of different materials processing, irradiation conditions and burnup.

3. FUEL

3.1. Uranium dioxide (UO_2) and doped UO_2

UO_2, with or without minor dopant additions is the fuel of choice for current LWRs. It was chosen because it has good resistance to oxidation in water and a high melting point. The main drawback is its low thermal conductivity which further decreases with temperature and irradiation and results in high thermal gradients and fragmentation of the fuel within the reactor. The behaviour of UO_2 in storage and disposal is relatively benign, in that it oxidises/corrodes slowly in air at low temperatures and in common groundwaters. When oxidised, UO_2 expands initially by accommodation of interstitial oxygen up to a stoichiometry of U_4O_9 then to U_3O_8 with a change of atomic structure. The volume expansion leads to a friable (dispersible) oxide and can cause progressive failure of cladding. Therefore, inert gas is preferred for dry storage. Irradiated fuel initially oxidises more rapidly than UO_2, however the presence of fission products tends to stabilise the U_4O_9 structure leading to a delay on subsequent oxidation [21]. Accelerated testing of UO_2 and MOX fuels has shown that volume expansion due to helium accumulation in the fuel matrix saturates at around 2%. At this level there is no concern over the integrity of Zr alloy or steel-clad fuels during long term storage. Some testing to assess the long term swelling behaviour of other fuel matrices would be recommended if such fuels are selected for deployment to confirm both the rate of accumulation and saturation levels.

The addition of small quantities of dopants to the UO_2 matrix can assist densification and grain growth, with consequential lower fission gas release and better resistance to fuel wash out [22]. Fuels of this type are already commercial products, such as the Westinghouse ADOPT™ fuel. Further work on doped fuels is underway as part of an EU funded Horizon 2020 project, Modern Spent Fuel Dissolution and Chemistry in Failed Container Conditions (DisCo) [23]. There is also interest in more advanced UO_2 fuels where the goal is to increase the thermal conductivity, for example by creating a network of higher thermal conductivity material within the microstructure [24]. These fuels, which have benefits of reducing peak pellet centreline temperatures enabling more efficient heat removal, require higher quantities of dopant materials (typically 10 wt%). Spent fuel management of these advanced fuels has yet to be assessed but is not expected to be significantly inferior to UO_2.

3.2. Uranium Nitride (UN)

Uranium Nitride (UN) is being considered as a candidate ATF due to its superior thermal properties and higher uranium density when compared to UO_2. Reactivity with water and oxygen is the main concern for UN since it is known to be significantly inferior to UO_2, which could have implications for spent fuel storage and disposal. A number of studies of the oxidation and hydrolysis of UN have been undertaken on both solid and powdered materials [25,26]. The onset of rapid oxidation (ignition) of powders typically occurs around 300°C in air and 340°C in water vapour but depends greatly on the particle size. Solid materials are relatively stable at room temperature due to the formation of epitaxial layers of U_2N_3 and UO_2 which afford protection [26]. Within the temperature range encountered during fuel drying and storage, inert atmosphere conditions would need to be maintained. It is worth noting that the reactivity of UN to water is also a concern during reactor operation, due to the potential for fuel pin failures. Efforts are underway to improve the water tolerance of UN, for example by forming a composite with another material [27]. Hence there is a reasonable expectation that any fuel so developed will also perform adequately in the conditions encountered during wet or dry storage. Testing of these modified fuel forms will need to be undertaken in representative conditions over sufficient time periods to underpin this assumption.

Carbon-14 is a known fission product when using standard nitrogen (99.63% N-14). This can be volatised which would cause the release of a radioactive gas. This could be a significant challenge for storage and disposal facilities. The proposed solution is to enrich the nitrogen in N-15, however it is not yet clear the level of enrichment that would be required to keep C-14 production to acceptable levels and enrichment processes are not yet economic on a large scale.

3.3. Uranium Silicide (U_3Si_2)

Uranium Silicide (U_3Si_2) is also being considered as an ATF candidate fuel for the same reason as UN (improved thermal conductivity and uranium density). Although the increase in uranium density is lower than for UN (17% compared to 40%), there are no isotopic concerns. Like UN, U_3Si_2 is known to oxidise in both water and oxygen, although testing under pressurised water conditions shows notably better performance than UN [28]. Ignition of bulk U_3Si_2 in air typically occurs at around 400°C, and in steam rapid pulverisation occurs at around 460–480°C [29]. The pulverisation in steam is thought to be associated with the formation of a uranium silicide hydride ($U_3Si_2H_{1.8}$) phase, with similarities to the way that hydrogen can pulverise bulk uranium [29]. At lower temperatures in water, cast bars of U_3Si_2 exhibited little evidence of reaction in water over 16 days at 100°C and 4 days at 200°C [30]. Longer term testing has not yet been performed. As with UN there is currently on-going research aimed at improving the water tolerance of U_3Si_2 in reactor conditions, one example being the addition of dopants [31]. Promising candidate compositions will also need to undergo long term testing at low temperatures of both unirradiated and irradiated materials to underpin their safe storage and disposal.

4. CONCLUSIONS

Significant international research is being undertaken to develop ATF for LWRs that could provide greater resilience from the fuel in the event of a severe accident. A range of different fuel and cladding concepts are under development which might be described as evolutionary or revolutionary by the extent to which they deviate from the current fuel. This paper has provided an initial assessment of spent fuel management considerations for a number of these concepts that are being actively researched. Spent fuel management has received little consideration to date, but this is expected to change as concepts progress to licensing. Of the concepts assessed the following conclusions can be made:

- Evolutionary concepts, such as coated Zr alloys and doped UO_2 fuels, which are closest to commercial deployment, will benefit from current experience with standard UO_2/Zr alloy fuel, and although some testing is likely to be required, their behaviour is expected to be bounded by the current fuel;
- Advanced iron alloy claddings can be expected to exhibit lower general corrosion rates than Zr alloys. These materials are however prone to localized corrosion resulting from radiation induced sensitization. Evidence from the early use of austenitic stainless steels in LWRs indicates that these effects should not be an issue, although corrosion inhibitors have been successfully used during pond storage and could be adopted if required;
- SiC/SiC composites are expected to be exceedingly durable cladding materials in long term storage, although data underpinning this assumption will be required. Differences in the quality of SiC/SiC composites with processing and joining technologies will need to be accounted for and the potential for the evolution of carbon containing species assessed;
- Both UN and U_3Si_2 fuels show a greater propensity for corrosion than UO_2 in both air and water. Improvements in water tolerance are being sought to ensure their suitability for use in LWRs and it is these composite or doped fuels that will require testing at low temperatures relevant to storage and disposal conditions.

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Paper ID#172
PARTITIONING OF HIGH LEVEL LIQUID WASTE IN CHINA

C. XU
Institute of Nuclear and New Energy Technology, Tsinghua University
Beijing, China
Email: xuchao@tsinghua.edu.cn

J. CHEN
Institute of Nuclear and New Energy Technology, Tsinghua University
Beijing, China
Email: jingxia@tsinghua.edu.cn

G. YE
Institute of Nuclear and New Energy Technology, Tsinghua University
Beijing, China

X. LIU
Institute of Nuclear and New Energy Technology, Tsinghua University
Beijing, China

Abstract

The main development of high level liquid waste (HLLW) partitioning in China was briefly reviewed. Chinese high level liquid waste has been stored for several decades. How to manage the historic HLLW is a serious problem. A total partitioning process has been developed at Tsinghua University. The process consists of the following three extraction cycles: actinides removal by TRPO extraction, Sr-90 removal by dicyclohexyl-18-crown-6-ether extraction and Cs-137 removal by calix [4] arene extraction. Based on the partitioning process, the hot test facility including 72-stage miniature centrifugal contactors was set up in the hot cell. About 300Ci HLLW was continuously partitioned within 160 hours through this facility. After 120-hour operation, 30% TRPO-kerosene was recycled without any treatment. The average values of decontamination factor were determined to be more than 3×10^3 for α activity and more than 10^4 for Sr-90/Cs-137, respectively. These results demonstrate that Chinese historic HLLW can be transferred into non-α and intermediate/low level waste by the total partitioning process.

On the other hand, developing nuclear energy has been chosen as one of important direction for energy resources in China. The development of nuclear energy cannot be separated from the support of the nuclear fuel cycle. How to effectively manage the nuclear fuel cycle, especially high level waste from the commercial reprocessing plant, to support the sustainable development of nuclear energy is an intractable problem in China. Partitioning of HLLW provides an option to reduce the high level waste which needs be disposed in the geological repository. The research work on partitioning of commercial HLLW is under process in China.

1. PARTITIONING OF HLLW

1.1. Management of high level waste

High level liquid waste (HLLW) comes from the reprocessing of spent nuclear fuel, i.e. PUREX raffinate. The safe treatment and disposal of HLLW is very important in the management of nuclear waste and crucial to the sustainable development of nuclear energy. The partitioning and transmutation (P&T) of long-lived radionuclides such as minor actinides from HLLW is an attractive method to reduce the long term risk of high level waste.

However, there is a very long way to go in the field of long-lived nuclides transmutation. The near-term objective of HLLW management is to reduce the long term risk in disposal and the treatment cost. A supposed programme in Fig. 1 is to separate HLLW into α waste, high radioactive waste and non-α intermediate level waste (ILW). So, the main purpose of partitioning is to realize the highly efficient removal of actinides and Sr-90/Cs-137 from HLLW.

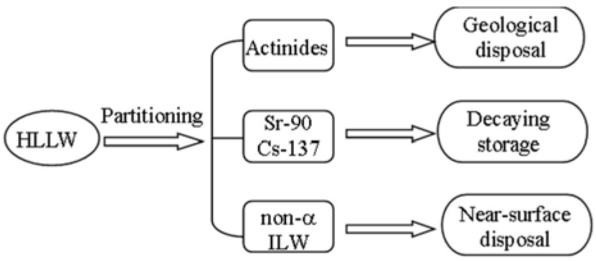

FIG. 1. Programme of HLLW partitioning.

1.2. Partitioning technology

Partitioning of HLLW, especially for actinides separation, has been studied for several decades. Some partitioning processes, such as TRUEX process [1], DIMAEX process [2] and TRPO process [3], were developed and have continuous improvement.

Recently, new trends appeared in the field of HLLW partitioning. Some new ligands were developed for the aqueous process. For example, the ARTIST (Amide-based Radio-resources Treatment with Interim Storage of Transuranics) process was developed by the Japan Atomic Energy Agency (JAEA) [4], in which TODGA (Tetraoctyl-diglycol-amide) and DHOA (di-n-hexyl-octanamide) were used as the organic ligands. DOTGA process is worth of attention because European scientists also show very strong interest [5]. Pyro-process was paid a lot of attention recently. Most research focused on the electrorefining technology [6].

Some new alternative methods to conventional solvent extraction methods were presented such as supported liquid membranes, ion exchange resins and extraction chromatography [7]. The extraction chromatographic procedure employs solid extractant which is prepared by impregnating ligands with inert support materials such as poly(4-vinylpyridine), silica gel, and so on.

Another trend is to combine the partitioning with modified PUREX process to integrate the reprocessing. UREX+ process [8], one of the key attributes in US-DOE's AFCI plan should be the typical representative.

The removal of Sr-90 and Cs-137 is also important to reduce the heat release of high level waste. The popular technology for Sr-90 and Cs-137 removal is crown-based solvent extraction, for example dicyclohexyl-18-crown-6 as extractant for Sr-90 and calixarene as extractant for Cs-137 [9].

2. PARTITIONING RESEARCH IN CHINA

2.1. Partitioning process

In 1980s, actinides partitioning process from commercial HLLW was developed based on trialkyl(C_6-C_8) phosphine oxide (TRPO) extraction [10]. Figure 2 shows TRPO process. Actinides are extracted by 30% TRPO-kerosene. Am(Cm), Np+Pu and U are stripped by nitric acid, oxalic acid and Na_2CO_3, respectively. TRPO process was tested using German WAK HLLW with good partitioning efficiency in 1992 [11].

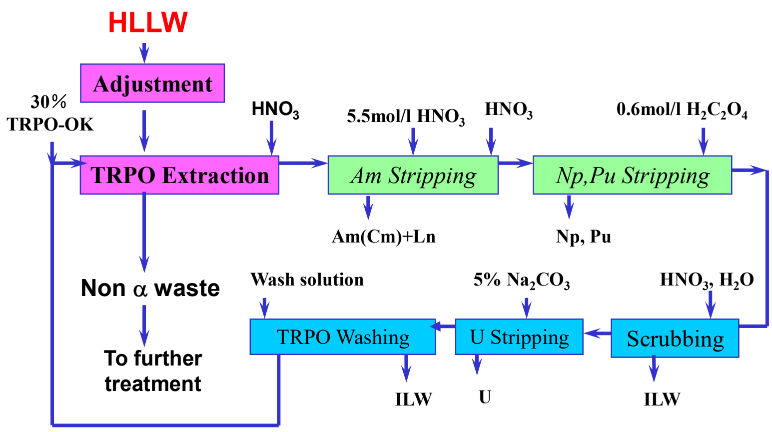

FIG. 2. Flowsheet of TRPO process.

From 1990, TRPO process was applied to separate Chinese highly saline HLLW. Meanwhile, the Sr and Cs separation technology from highly saline HLLW was studied. The crown ether was selected as the extractant for Sr separation and potassium titanium hexacyanoferrate (II) as inorganic ion exchanger for Cs removal. Around 2000, a total (actinides, Sr and Cs) partitioning process was presented [12]. In 2005, a pilot facility was built to test processes for actinides, Sr and Cs partitioning with Nd and Zr simulating Am and Pu, respectively. In 72-hour pilot test, the experimental decontamination factors (defined as the content ratio of metal in HLLW to in raffinate) are >3000 for U, >500 for Nd, >1000 for Zr, ~160 for Sr, ~100 for Cs, respectively [13], which demonstrated the feasibility and reliability of the total partitioning process and the used separation equipment.

After pilot test, the disadvantage of ion exchange operation for Cs removal appeared because of more complicated controlling under radioactive condition compared with solvent extraction operation. Calixarene was introduced to extract Cs from HLLW instead of ion exchanger. Finally, the improved total partitioning process consists of three extraction cycles (Fig. 3): actinides separation by TRPO extraction, Sr separation by crown ether extraction and Cs separation by calixarene extraction.

2.2. Hot test

A 5h hot test for TRPO process, Sr removal by crown ether extraction and Cs removal by ion exchange was finished in 1996. In order to test the modified process with three extraction cycles, a 160 h hot test for the total partitioning process based on solvent extraction was carried out in 2009. 72-stage miniature centrifugal contactors were employed in a hot cell. About 300 Ci Chinese highly saline HLLW was partitioned. Figure 4 shows the partitioning result. The average decontamination factors are more than 3000 for α activity, and more than 10 000 for Sr-90 and Cs-137, respectively [14]. These results completely meet the requirement to change high level waste into non-α and intermediate level waste.

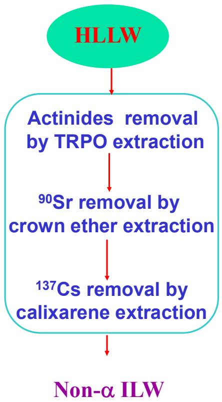

FIG. 3. Total partitioning process of three extraction cycles.

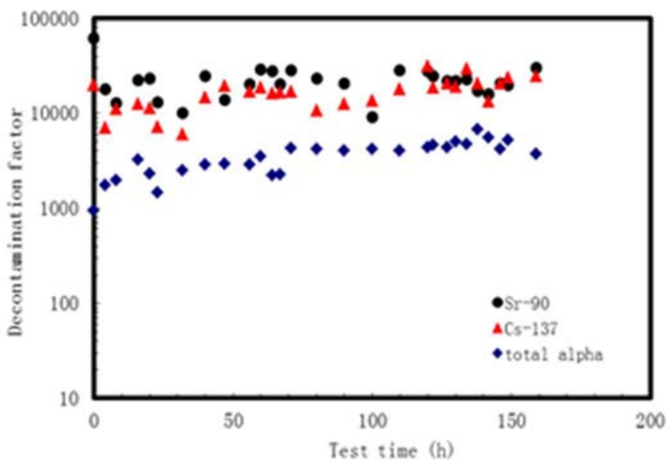

FIG. 4. Decontamination factors of α activity, Sr-90 and Cs-137 in the hot test.

3. PROSPECT

In the future, we have two tasks in HLLW partitioning. On the one hand, a pilot hot test facility should be built to demonstrate our total partitioning technology for highly saline HLLW partitioning.

On the other hand, the growing electricity demand is stimulating the nuclear power expansion in China. With the discharge of spent fuel, the commercial reprocessing plant rose to an agenda. The treatment of commercial HLLW is concerning subsequently. Chinese scientists agree that P&T-based advanced nuclear fuel cycle is the best option to solve the problem of high level waste. So, the further study on the partitioning of commercial HLLW and integrate the PUREX process and partitioning process are in progress. Because of the difference between the highly saline HLLW and the commercial HLLW, we must adjust the management objective of HLLW to simplify the process and reduce the cost. Figure 5 shows the simplified concept of HLLW management. Actinides and fission products are just separated from each other. Actinides separation from HLLW is the preferred alternative considering the transmutation of TRUs in the future. However, high separation factor of actinides must be achieved to ensure change the raffinate fission products into non-α waste. This is the main challenge for TRPO process in the future.

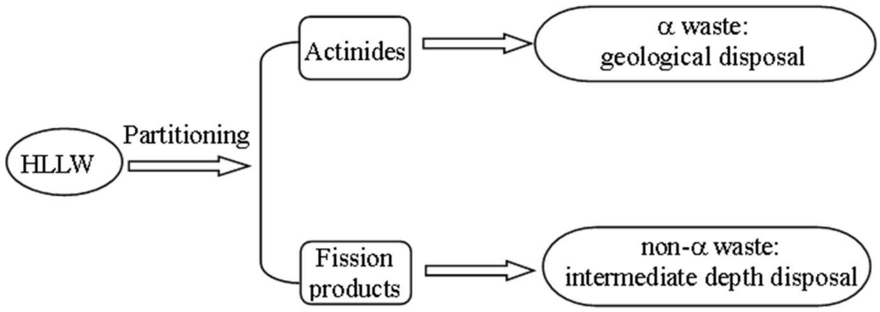

FIG. 5. The simplified concept of HLLW management.

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3.6. TRACK 6 – DISPOSAL

Overview prepared by H. Zaccai (World Nuclear Association) and T. Tate (United States of America), **Track Leaders**

The session included a very diverse set of presentations which highlights various challenges in spent fuel and HLW disposal as well as multiple potential options to resolve them.

The Environmentally Safe Disposal of Radioactive Materials (EDRAM) representative made an overview of the issues related to disposal: siting, multinational approaches, cost and knowledge management.

Final disposal of spent fuel and HLW has been analysed in terms of multiple nuclear fuel cycle options. The French final disposal programme has been presented by ANDRA as being an on-going industrial project in its design phase aiming at starting the inactive phase around 2030.

Possible financing approaches of a multinational repository have been presented by the International Framework for Nuclear Energy Cooperation (IFNEC). The effort will be pursued with the view to engage service providers and potential customers.

One theme particularly highlighted by the Nuclear Waste Management Organization (NWMO) in Canada and also by Finland was the vital and essential role that local stakeholders must play, for any future disposal programme should take these socio-political factors into account.

Session 6.1: Disposal

Session Chairs: H. Zaccai (World Nuclear Association) and J. Home (United Kingdom)

Session 6.1 comprised of six presentations, two from International organizations, EDRAM and IFNEC, one from USA, one from France, one from Finland and one from Canada.

- **Paper ID#31 *(Invited)* by S. Kondo (EDRAM)** gave an overview of EDRAM group that works with several partner countries to develop programmes for HLW disposal. EDRAM suggests a proportionate approach which can optimize the implementation of radioactive waste disposal. Mr. Kondo also highlighted the role that international organizations like the IAEA and OECD Nuclear Energy Agency (NEA) should play in supporting a Multinational Repository.
- **Paper ID#185 *(Invited)* by P. Swift (USA)** provided insights from published safety assessments for disposal of spent nuclear fuel and HLW, suggesting that modifications to waste forms from potential advanced fuel cycles are not essential for demonstrating safe long term performance of repositories.
  Modifications that reduce the thermal power of the waste or that reduce waste volume without increasing thermal loading have potential to allow more efficient use of DGR. Due to the relatively higher mobility of the long lived fission product I-129 in most disposal system environments, changes in the radionuclide inventory of waste forms from the potential recovery and reuse of fissile material contained in spent fuel are unlikely to have a significant impact on the estimates of long term performance for most disposal concepts (in the absence of disruptions that expose the waste directly to the biosphere such as human intrusions).
  Waste form modifications for durability have the potential to improve estimated peak dose performance of repositories only if the modified waste-form lifetime becomes relatively long compared to the geosphere transport time, and/or approaches the period of performance, e.g., on the order of hundreds of thousands of years.
- **Paper ID#159 by J.M Hoorelbeke (France)** stated that ANDRA is preparing an application for the creation of a geological disposal facility in Eastern France. An underground research laboratory has been in operation since 2000. The project aims at disposing vitrified HLW produced from the reprocessing of spent fuel, as well as a range of intermediate level waste (ILW), including metallic parts of the fuel assemblies separated during reprocessing and various operational wastes. Cigéo design capacity covers existing HLW and ILW in France as well as waste which will be produced in the next decades by the operation and decommissioning of all existing French nuclear facilities. A total of 100-year operational period is envisaged. By law, the Cigéo project is designed consistently with reversibility requirements. Within this framework the very first period of the project will be a pilot industrial phase expected to start in 2030. Provisions are made so that next generations can adapt the operational process to accommodate potential changes in the French spent fuel and waste management policy and strategy.
- **Paper ID#120 *(Invited)* by T. Žagar (IFNEC)** stated that the development of a DGR involves high fixed costs that carry an associated economy of scale. A DGR with a capacity of 10 000 metric tonnes can cost little more than one designed to dispose of 5000 metric tonnes. This means that smaller nuclear programs could benefit greatly from the opportunity to participate in a Multinational Repository (MNR). That is the reason why IFNEC has launched an initiative of an MNR concept to provide a shared solution to the challenges of spent fuel and HLW disposal. The concept involves a

service provider country developing a geologic repository and accepting spent fuel from several customer countries. Recent developments regarding the identification of financing approaches for an MNR have been observed among different fora and are presented. IFNEC's paper focuses mainly on results of recent work done and presents four possible approaches:
  - Approach 1: It is clearly challenging to finance one MNR but may be easier to finance several;
  - Approach 2: Two Approaches: government lead with and without customer investment;
  - Approach 3: Sell shares in the repository project with return of investment coming from fees collected during operation;
  - Approach 4: Financing with a staged interim storage/repository approach.
- **Paper ID#106 by M. Kojo (Finland)** provided an interesting comparison of two geological disposal programmes: Eurajoki in Finland and Osthammer in Sweden. Both programmes focused on improving local stakeholder relations as top-down approaches had previously been shown to be ineffective. Local municipalities were given funding to carry out their own reviews and given a veto vote on the programmes. Osthammer adopted an active role in the host community while Eurajoki took more of a 'trusted bystander position'.
- **Paper ID#130 *(Invited)* by L. Frizzell (Canada)** spoke about how NWMO encouraged communities in Canada to volunteer to host the Canadian repository through outreach and education. This approach was rewarded as 22 communities volunteered and NWMO is currently assessing the final five potential sites. She highlighted that host communities are a vital resource of local information and with respectful engagement will be central to the success of the repository.

Session 6.2: Disposal

Session Chairs: T. Tate (USA) and M. Kari (Finland)

Session 6.2 comprised of five presentations, one from IAEA, one from Sweden, one from Finland, one from Russia Federation, and one from United Kingdom.

- **Paper ID#195 by I. Tsvetkov (IAEA)** explored the issues of safeguards by design. IAEA safeguards are intended to verify that nuclear material is not diverted from its supposed use. With safeguards by design it is possible to avoid costly and time-consuming redesigns and enhance efficiency. The idea is to incorporate safeguards infrastructure measures into facility design in the planning phase with the representation of all the stakeholders to find the optimal way of integration. Final disposal facilities in Finland were raised as examples of application of safeguards by design.
- **Paper ID#150 *(Invited)* by A. Sjöland (Sweden)** presented the important parameters involved in the characterization of spent nuclear fuel for transportation and disposal. The presentation discussed the parameters to characterize are decay heat, criticality, radiation doses, nuclide inventory, and safeguards. Decay heat is important and often the driving parameter in all parts of the back end system. Factors considered for decay heat management in a final repository are the passive cooling system, the importance of canister optimization in the design, and the importance of the fundamental parameters in codes such as SCALE. Calorimetry has the potential to provide accurate measurements of up to 2–3% but requires long measurement times. International collaboration on blind tests of decay power predictions has occurred consisted of

around 25 groups and organizations. The blind tests involved comparing predicted value and calorimetric measurements of 5 Swedish fuel elements. The importance of accurate radiation dose determinations and challenges with safeguards were discussed. Characterization of uncertainties, fuel information, and fuel integrity were also discussed.

- **Paper ID#117 by P. Mäenalanen (Finland)** presented on the regulatory control of nuclear facility licensing steps and construction license review in Finland. It was noted that although there was a lot of experience regulating nuclear facilities, in the case of final disposal repository, regulatory control had to be adapted for underground facility. The experience from the oversight of the Onkalo underground test facility has been valuable in this and made stepwise licensing useful. The operating license application for the repository is planned to be submitted by the end of 2022, and part of it will be soon pre-reviewed by the Radiation and Nuclear Safety Authority (STUK).
- **Paper ID#137 by P. Blokhin (Russian Federation)** presented the estimation of radionuclide inventories on high level radioactive waste. The presentation discussed validation and benchmark testing of TRACT code radionuclide inventory against decay heat measurements. The comparison concluded that there was good agreement in the results, but there were some discrepancies discovered in the gamma characterization. Data of various measured and calculated isotope comparisons were presented to demonstrate the ability of the code to predict uranium oxide fuel radionuclide inventories. The comparisons supported the conclusion that the radionuclide inventories for WWER-440 reactor fuel can be well predicted by the TRACT code.
- **Paper ID#54** ***(Young Generation Challenge Winner)*** **by J. Home (United Kingdom)** presented the human interface challenges for long term use and disposal of spent nuclear fuel. Nuclear wastes remain radioactive for 100 000 years while a geologic disposal facility must be designed to prevent human interaction with the waste until it is safe. A consideration for challenges in the material selection, messages, use of symbols, design, and of security were presented.

Paper ID#31
RECENT PROGRESS IN GEOLOGICAL DISPOSAL OF RADIOACTIVE WASTES AND STRATEGIC ISSUES TO BE DEALT WITH IN THE PROCESS: EDRAM'S PERSPECTIVE

S. KONDO
Chair of EDRAM / Nuclear Waste Management Organization of Japan (NUMO)
Tokyo, Japan
Email: skondo1@numo.or.jp

Abstract

Various issues of deep geological disposal of radioactive waste, including alternatives to geological disposal, multinational approaches and costing / financing aspects are being discussed in the public and political sphere in a recurrent manner. EDRAM believes, as a group of senior executives from national agencies for implementing radioactive waste disposal in their respective countries, that drawing on international expertise, experience and collaboration is of great value and leads to better solutions for the safe implementation of radioactive waste disposal. Based on this belief EDRAM discusses strategic issues and technical and management matters, with a view to benchmarking and establishing best practices, develops a common understanding of waste management issues among implementers and positions thereof and coordinates actions in relation to international organisations. EDRAM continually exchanges information on these matters within the group and with international organizations and understands differences and commonalities among them deeply in order to be able to explain them to its stakeholders. In the paper summarized are some of major recent outputs from this discussion.

1. INTRODUCTION

Development of solutions for long term management of spent nuclear fuel (SF) and high level radioactive waste (HLW) should proceed irrespective of the future of nuclear power generation, although volumes of SF and HLW produced are small and they are safely stored on an interim basis and can be continued to be safely stored using current practices for many decades. Many countries have promoted research and development (R&D) programs on long term management of them and concluded that their disposal in a deep-mined, geological repository (DGR), i. e. their geological disposal is technically safe and feasible.

After establishing a national strategy for the geological disposal and an agency that is responsible for the implementation of the strategy, some countries are now promoting the development of geological repository, having identified a potential site for such a repository. In Finland, construction license was granted for the first deep geological repository for SF [1], in Sweden license application was submitted [2] and in France [3], the application is to be submitted in this year. In Canada [4] and Switzerland [5], the siting process is ongoing, and its definition is advanced in Germany [6], Japan [7] and UK [8], though the progress in these countries take tortuous routes from time to time.

International Association for Environmentally Safe Disposal of Radioactive Materials (EDRAM) [9] is a non-profit association established in 1998 as a forum to promote exchange of knowledge, experience and information among senior executives from national agencies for implementing geological disposal of radioactive waste in their respective countries. Participation includes 12 organisations from Europe, North America and Japan.

EDRAM recognizes that the International Atomic Energy Agency (IAEA) has played important roles of not only verifying that States are honouring their obligations to use nuclear material and technology only for peaceful purposes, but also supporting member States by helping to build national confidence in radioactive waste management activities through the publication of technical information including safety-related standards and guidelines and promoting effective cooperation and experience sharing among member States in various meetings and through review missions.

From global perspective, however, various issues related to radioactive waste management are being discussed in the public and political sphere in a recurrent manner, in particular, alternatives to geological disposal, multinational approaches and costing / financing aspects. Furthermore, as we are living in interdependent society, something happening in one country on waste management and, in particular, on deep geological disposal, has an immediate impact on other countries.

EDRAM believes, therefore, that we all need to continually exchange information on these matters between us and with international organizations so as to understand differences and commonalities among them deeply in order to be able to explain them to our stakeholders. There is a strong need for clear and coherent messages – in particular from international organisations – that will support, or in any case not adversely affect, advanced and future disposal projects.

Keeping this point in mind and believing that drawing on international expertise, experience and collaboration is of great value and leads to better solutions for the safe implementation of radioactive waste disposal, the EDRAM discusses strategic issues and technical and management matters, with a view to benchmarking and establishing best practices, develops a common understanding of waste management issues among implementers and positions thereof and coordinates actions in relation to international organisations.

In the following, summarized are some of major recent outputs from this discussion (Note: The current presentation is prepared based on the products of EDRAM, but the views presented here are those of the author and do not represent necessarily the views of the EDRAM).

2. EDRAM'S POSITIONS AND VIEWS

EDRAM recognizes that independently of future developments in nuclear energy, nuclear waste exists and must be managed in a safe and sustainable manner – now and in the very long term. For this management, strategies are needed from cradle to grave, i.e. from waste generation to disposal – for all types of nuclear waste. A range of options regarding depth of disposal exists (from near-surface to deep geological disposal) and must be considered in the context of each national framework, based on the principle that:

— The burdens and responsibility for taking care of radioactive waste should not be passed on to future generations;
— Radioactive waste management is a societal, as well as a technical issue;
— There is a need for flexibility, as well as for open and ethical involvement of stakeholders in decision making.

EDRAM has developed position and views from time to time based on the discussion of strategic issues among implementers, where appropriate, for communicating its common view with international communities, whilst recognizing the value of open discussions with national governments and regulators. Included in the present paper are views on siting of DGR, community benefits, multilateral approaches, a proportionate approach to radioactive waste disposal, partitioning and transmutation on the long term management of radioactive waste, and cost for geological repository projects.

2.1. Siting of DGR

Based on our learning and experiences from geological disposal siting processes, we have discovered that a proposed repository site must not only be technically suitable but also be socially acceptable. Social acceptability could not be secured by relying on the authority of science and the power of government. Achieving a sustainable level of social acceptability requires, at a minimum, a transparent process that respects the views of interested and affected parties, that appreciates the authenticity of those beliefs, and through which share with the public about the information on the assessment of geological suitability, decision making processes with right of withdrawal, community benefits and potential socio-economic and environmental effects. Successful projects now refer to forging partnerships with local communities, implying a more equitable and enduring relationship, to the benefit of safe implementation.

We also recognize that siting is complex and multi-dimensional; approaches will differ from country to country: implementers are pursuing win-win situations with communities that are considering acceptance of waste repository. Recognizing that siting processes are challenging, we must be prepared to adjust ourselves according to regional circumstances and variations in societal requirements.

2.2. Community benefits

EDRAM commissioned Anne Bergmans of University of Antwerp to prepare a report of which purpose was to establish an overview of community benefits that were made available for communities hosting radioactive waste facilities in EDRAM member states [10]. The report pointed out that:

- Where the siting of radioactive waste management disposal facilities is concerned, socioeconomic community benefits generally form a substantial part of facility siting efforts and good community relations in EDRAM member states.
- Although all implementers stress the impact of the facility on the local economy as a benefit in itself, in terms of additional employment, local procurement and potential spin-offs, there is growing recognition that communities willing to fulfil an essential service to the nation by hosting a final repository of radioactive waste are entitled to receive added-value measures to develop their social and economic wellbeing.
- These additional benefits come in various shapes but tend to fall under either one of the following five categories: (1) additional investments in local infrastructure, (2) additional local activity, (3) specific subsidies and grants, (4) offering support in the form of training and logistics, or (5) setting up community funds for local development. Only in a few countries, a specific tax or a particular tax-rate applies for nuclear installations, including radioactive waste management facilities.
- The limited contextual information gathered for this study indicates that context matters and that the nature, dimension and scope of community benefits are predominantly determined by the social and political context, as well as the needs and requirements of the host communities through negotiations between the implementer and the host community. These negotiations in all cases form, in one way or the other, an integral – albeit not necessarily formal – part of facility siting procedures, running in parallel with site investigations and feasibility studies. In most cases, negotiations on community benefits are concluded before licensing.
- The processes that concerned parties go through to negotiate benefits and define added value are crucial in determining whether or not such benefits (quantifiable or not) are seen as appropriate by all concerned. Through these processes, relationships are built which have an impact on the perception of appropriateness of the agreed benefits for that specific situation.

2.3. Multinational approaches

It is generally accepted that the ultimate responsibility for ensuring the safety of spent fuel and radioactive waste rests with the government of the country in which it was generated. This does not mean, however, that the fulfilment of national obligations through collaboration with other countries should be precluded. The Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management states "that, in certain circumstances, safe and efficient management of spent fuel and radioactive waste might be fostered through agreements among Contracting Parties to use facilities in one of them for the benefit of the other Parties, particularly where waste originates from joint projects". With this recognition, the IAEA published in 2016 a report [11] that described a phased approach and indicated the decision processes to be followed by partners in the multinational project to realize a shared disposal facility, both within a national context and in the scope of the joint endeavour, touching on a wide range of legal and institutional aspects, including the contractual obligations among partners; economic and financial arrangements; liabilities; nuclear security; regulatory and legislative aspects; waste transportation arrangements and social matters. The uncertainties and risks involved in the implementation of a multinational repository are also addressed.

Members of EDRAM are implementers of geological disposal of higher radioactive waste in their respective countries established in accordance with their national policy and responsible to realize DGR(s), promoting research and development necessary for attaining their mission and selection of a repository site based on a strategy for the management of their radioactive waste specified by the government. Some of them have established authentic interactions with communities embracing trust and transparency as a critically important organizational priority and have been successful in forming strong bonds of trust with local populations. Therefore, EDRAM believes, though recognizing the involvement in a multinational repository project should be as one of the options in a national policy and strategy, that a successful implementation within the next 10–20 years of some advanced national programmes should be a top priority as it will be a showcase for institutional and technological success in the management of higher radioactive waste.

2.4. A proportionate approach to radioactive waste disposal

As far as the long term management of higher activity radioactive wastes is concerned, geological disposal is the only acceptable option, and will be required regardless of the introduction of treatment options such as partitioning and transmutation. However, a number of countries have identified a proportion of their waste inventory that, whilst being unsuitable for surface disposal, may not require geological disposal at depths greater than 200 to 300 meters.

EDRAM accepts that such an approach to the management of radioactive wastes is appropriate subject to the following considerations:

- Any disposal concept should be based on provision of safety functions appropriate to the degree of isolation and containment required for the radioactive waste over suitable time scales.
- When defining the disposal concept, a number of means such as the site characteristics, the waste forms or the engineered barriers system can be called on in order to adequately satisfy the above-mentioned requirements.
- In particular the selected depth of any disposal facility contributes to the degree and duration of isolation and of protection from surface erosion due to effects such as glaciation.
- The depth of the disposal facility also plays an important role in reducing the likelihood of human intrusion.
- Existing surface disposal facilities may not provide the safety functions needed for long-lived radioactive waste. By extending the depth of facilities below the ground level as needed, the degree and/or duration of isolation, protection from natural surface processes and potentially containment can be enhanced. Such facilities could then be suitable for the disposal of material that would present a relatively low hazard, with regard to their radiological inventory, such as irradiated graphite, some operational waste and decommissioning wastes.
- Though any approach will need to demonstrate compliance with the applicable safety standards, this proportionate approach can optimize the implementation of radioactive waste disposal solutions.

2.5. Partitioning and transmutation on the long term management of HLW

A number of countries are publishing study reports on the alternative management strategies for long term management of HLW including partitioning and transmutation as in the case of Finland [12], France [13], European Union [14], Sweden [15] and United Kingdom [16], often within a framework of environmental impact assessment. Based on the arguments presented in these reports, EDRAM draws the following conclusions:

- Minor actinide partitioning and transmutation is technically feasible, but there are considerable technological uncertainties to address, and major R&D investment would be needed both by the international community and by individual countries wishing to implement it.
- Transuranic elements contribute largely to the radiotoxicity of HLW, but they have only a minor impact in terms of long term radiological risk. Therefore, minor actinide partitioning and transmutation does not displace the need for geological disposal: the transmutation option must be considered as a possible complement, but not a substitute for waste disposal.
- The long term radiological risk results from a limited number of fission and activation products. Although there are scenarios where a strategy of separating minor actinides for recycle can reduce the geological disposal footprint, the benefit of transmutation on the radiological risk of geological repositories remains to be demonstrated, considering that advanced nuclear technologies used for the implementation of the strategy will also produce high level and long-lived wastes, which will have to be managed in the long term.
- Partitioning and transmutation can only be applied to elements contained in future waste; it is not realistic to consider such a process for vitrified high level waste produced to date.

- The transmutation of transuranic elements requires a major investment by the entire nuclear industry of both economic resources and research and development into new technologies and their industrialization.
- A transmutation system comprises not only fuel reprocessing and fuel production units, but also advanced (high flux) reactors or accelerator driven systems that have not yet been developed. The capacity of the separation technology to provide the high level of purity required for transmutation and the ability of such a system to stabilize and reduce the overall amount of transuranic elements are yet to be demonstrated.
- Regardless of the technologies applied, transmutation is a slow process; the stabilization of an inventory of transuranic elements will take decades. Furthermore, in the framework of the present-day technical transmutation approach, curium is no longer considered because of the associated difficulties. Maintaining curium in high level waste will significantly reduce the benefit of transmutation in terms of radiotoxicity.
- A partitioning and transmutation programme can only be justified by ambitious goals in terms of reducing the radiotoxic inventory of radioactive waste. Only a very good global efficiency of the transmutation system could allow reaching these goals.
- It is necessary to follow up national and international developments concerning the partitioning and transmutation technologies cautiously as the reduction in the radiotoxic inventory of waste may constitute a favorable element for the societal acceptance of the disposal facility, as well as for the reduction in its footprint by reducing the heat load of the waste.

2.6. Cost for geological repository projects

The correct assessment of future costs for geological repositories is an important aspect of radioactive waste management. Nuclear power generators have been and are collecting funds for future costs of managing the waste from their nuclear power generation from the users of nuclear power including the fees for the costs in the current price of electricity. As no geological repository for civil high level radioactive waste and spent fuel is in operation yet in the world, the future costs need to be estimated. As the time differences between revenue generation and future expenditures can extend to over a century, it is crucial to have a sound methodology for the estimation to minimize the risk of transferring financial liabilities to future generations.

EDRAM established a working group [17] and asked to define tools and basic guidelines for the comparison of cost assessments of geological repositories, reviewing how the responsible authorities/companies in ten EDRAM countries handle the management of radioactive waste issue, including how the economic and financial aspects of the process are dealt with and where the money comes from. The study showed that while national policy and legal frameworks for repositories was broadly similar in these countries, specific circumstances directly relevant to comprehensive cost estimation and evaluation of available financing schemes were very different. These differences include the responsibilities and obligations concerning these issues of the waste producers, the waste disposal implementers and the regulators.

Other important considerations impacting on costs are the repository design (characteristics of rocks and depth of repository), its disposal capacity, annual receiving capacity and planned operating life time. Important factors bearing on the choice of an appropriate financing scheme include the charging arrangements and the application of the 'polluter pays' principle for repository and for related waste management tasks including interim storage of waste to be disposed of and necessary R&D activities. Payment timing; fund management responsibilities and investment policy; and methodologies for estimating costs and considering contingency margins that should be added to reflect the maturity status of the projects and their associated technologies uncertainties over the full repository life cycle could impact on cost significantly, though vary from country to country.

Comparing cost estimation methodologies on an international level may be a means to achieve global improvement, making comparisons is, however, not an easy task as every country is responsible for managing its own waste and a multitude of first-of-a-kind projects on various scales are to be taking place in different geological environments and in different societies with different economies.

The activities of this working group are still in a trial and error phase: the group is trying to make a break-down of the cost estimations in a standardized costing matrix that compares significant scopes of the programme consistent with the phases of the programme, expecting that this break-down allows comparing the economic aspects of the different programmes. However, the cost data are distributed over the involved quantities of SF, HLW and ILW, according to keys based on dimensions data of geological repositories. This distribution is important, as some countries only dispose spent fuel, while some countries only dispose high level (vitrified) and intermediate level waste, and other countries co-dispose of SF, HLW and ILW. At present the working group concludes that a simplified comparison of cost assessments for geological repository projects is not possible or would produce distorted conclusions. A sound comparison implicates a deconstruction of the cost estimates based on a good knowledge of repository designs for each waste inventories.

2.7. Knowledge management

EDRAM recognizes that the IAEA has been developing several new initiatives of knowledge management (KM) focusing on supporting Member States in their efforts to transfer and preserve knowledge, exchange information, establish and support cooperative networks, and train the next generation of nuclear experts, as many of nuclear organizations in the world face with challenges due to loss or lack of experienced staff, as decisions affecting safety and performance of nuclear power plants and wider stakeholder acceptance thereof must be made using the best knowledge and information available. The basic approach of KM is to utilize information technology (IT) for accomplishing effective use of knowledge and information, including such knowledge that forms best practices and lessons learned. As such knowledge is the result of human and organizational creativity/learning, KM activity should include aspects of learning and creation of knowledge as well as knowledge sharing and communication.

Reviewing recent international initiatives in the field of knowledge management from the waste management organisation (WMO)’s perspective, EDRAM recognized the need for:

— Developing a collective capacity aiming at addressing the full scope of the knowledge ecosystem;
— Dialogue between WMOs to facilitate the collective long term learning and competences sourcing;
— Collectively managing knowledge by steering cognitive flows and setting-up processes for capitalizing knowledge (capture, retention, sharing, access, reuse and update within each phase) in order to ensure its transfer to the next phases of the projects.

EDRAM recently established a working group with a view to sharing experiences on developing and implementing KM strategies in these respects for preserving knowledge capitalized across five or more generations and used to inform major programme decisions.

3. CONCLUSION

Believing that international expertise, experience and collaboration are of great value and lead to better solutions for geological disposal of SF and HLW, EDRAM regularly exchanges information among us so as to deeply understand our differences and commonalities from technical and industrial points of view. And it prepares notes of our common understanding or position papers based on such discussions for communicating them with international organizations and international communities, taking opportunity of international gatherings such as the present conference. Topics taken up in this presentation were mainly those related to siting and social acceptance. But it should not be understood that those topics were central issues for the progress in the implementation of geological disposal. Implementation itself includes many challenges: to obtain permissions to construct and operate the repository from hosting community as well as regulatory body, promotion of design and engineering, project management, human resources development, knowledge management etc., which spread over a century. EDRAM will continue to communicate its views on these challenges with international communities as a representative of implementers’ community, giving broader presentation, when necessary.

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Paper ID#185
IMPACTS OF NUCLEAR FUEL CYCLE CHOICES ON PERMANENT DISPOSAL OF HIGH-ACTIVITY RADIOACTIVE WASTES

P. N. SWIFT
Sandia National Laboratories
Albuquerque, USA
Email: pnswift@sandia.gov

D. C. SASSANI
Sandia National Laboratories
Albuquerque, USA

Abstract

All options for generating power from nuclear energy generate radioactive waste products that will require permanent isolation from the biosphere. Choices made regarding nuclear fuel cycle options, including decisions for recovery and re-use of fissile material from irradiated fuel, have the potential to affect the waste stream characteristics such as mass, volume, radioactivity, and thermal power, but no options eliminate the need for robust isolation of wastes. Decades of experience has produced an international consensus that deep geological disposal is the preferred method for achieving permanent disposal. The paper reviews published results of safety assessments for deep geologic disposal concepts that have been proposed in the United States, Sweden, France, Switzerland, and other nations to provide insight into the waste form aspects that most affect the long term performance of repository systems. Disposal concepts considered include geologic repositories in multiple rock types in both saturated and unsaturated environments. Additionally, this work evaluates how repository performance may be affected by hypothetical waste form modifications from changes in fuel cycle choices.

1. INTRODUCTION

The recognition that deep geological disposal is the preferred option for achieving safe isolation of high-activity radioactive wastes dates from the 1950s [1], and many nations began researching specific disposal options in the 1980s. Mined repositories are in operation for some categories of transuranic and intermediate-level waste [2, 3] and a repository for spent nuclear is under construction in Finland [4]. Progress toward facility licensing has been slow elsewhere in the world, however [5].

This paper reviews published safety assessment results for five different disposal concepts: mined repositories in granite [4, 6, 7], argillite [8, 9], salt [2, 10, 11], volcanic tuff [12], and deep borehole disposal in crystalline rock [13, 14]. This work also provides insights on how specific changes to the waste form that might result from alternative fuel cycle choices might affect long term performance of each concept. Published analyses indicate that all five concepts have the potential to meet regulatory requirements and provide robust long term isolation for the existing waste forms from the existing fuel cycles in each program.

Hypothetical modifications to waste forms requiring deep geologic disposal that could result from alternative fuel cycles and that are considered here include:

— Reduction in the radionuclide inventory associated with recovery and re-use of fissile isotopes;
— Reduction in the radionuclide inventory associated with additional partitioning and transmutation of radioisotopes remaining after recovery and re-use of fissile isotopes;
— Reduction in the volume of waste associated with recovery and re-use of fissile isotopes;
— Reduction in the thermal power of the waste associated with recovery and re-use of fissile isotopes;
— Increases in the durability of the waste form in the repository environment resulting from further treatment of the wastes;
— Increases in the durability of spent nuclear fuel in the repository environment resulting from alternative fuel cycle choices.

2. BACKGROUND ON THE DISPOSAL CONCEPTS CONSIDERED IN THIS PAPER

The Radionuclide Inventory Requiring Geologic Disposal. Figure 1 shows the time-dependent radioactive content (activity) of typical light-water reactor fuel from the US following irradiation [12]. This example provides a useful representation of the radionuclides that require long term isolation from any fission-based fuel cycle that does not include recovery and re-use of fissile material. At early times the disposal inventory is dominated by the relatively short-lived fission products Sr-90 and Cs-137. As these isotopes decay over the first few hundred years, the total amount of radioactivity becomes dominated by the transuranic radionuclides Am-241, Pu-240, and Pu-239. After several hundred thousand years, the long-lived fission product Tc-99 becomes the dominant contributor to the total inventory, until the system becomes dominated by Np-237, Pu-242, and long lived isotopes of U and Th. These radionuclides are not necessarily the most important contributors to estimates of long term releases from repositories because the mobility (and immobility) of specific radionuclides within each disposal concept is a key control on long term releases.

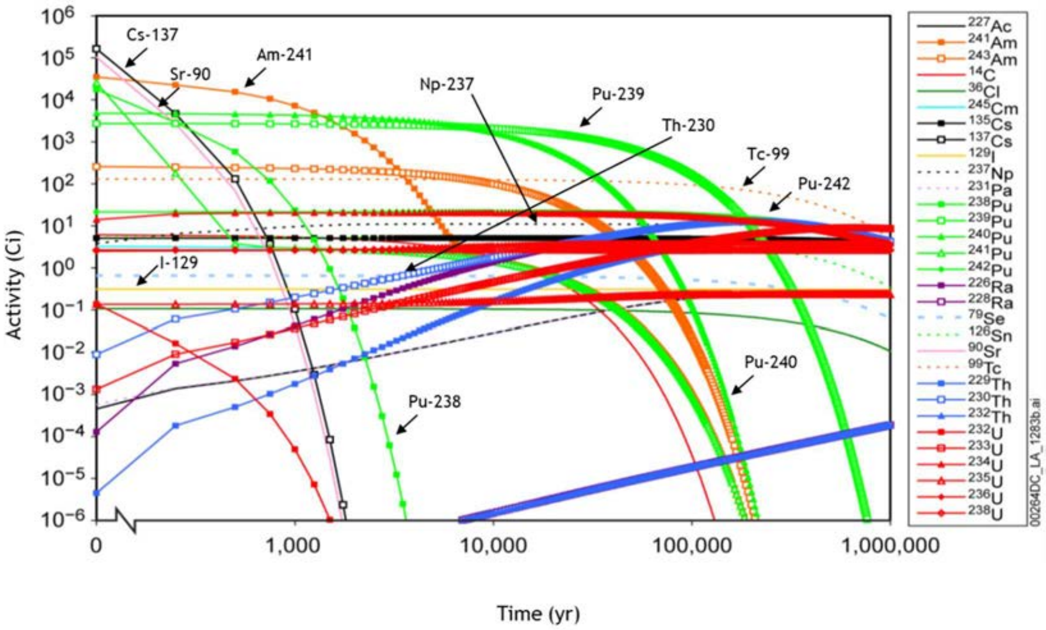

FIG. 1. Example of radioactive decay/ingrowth in irradiated spent nuclear fuel. Inventory activity is shown for a single representative waste package in the proposed Yucca Mountain repository, USA. Time is shown logarithmically in years after 2117. Source: [12], Figure 2.3.7-11.

Mined Repositories in Granitic or Crystalline Rock. Published analyses of the proposed repositories at Forsmark in Sweden [6, 15], Olkiluoto in Finland [4] and a generic site in Canada [7] provide representative examples of disposal in a mined repository in granite or granitic crystalline rock. The concept calls for emplacement of spent fuel in copper canisters in holes drilled in the floor or walls of a mined facility at a depth of several hundreds of meters in granitic crystalline rock. Groundwater at that depth is reducing, and primary barriers providing isolation of the waste include the low dissolution rate of uranium oxide (the primary component of spent fuel) in reducing groundwater, the stability of metallic copper in reducing groundwater, and the capability of a bentonite clay buffer emplaced around each canister to prevent advective groundwater flow in undetected fractures in the granite from reaching the canisters. When waste packages fail, radionuclide sorption in the bentonite and radionuclide precipitation in the reducing groundwater will lower the magnitude of releases that may reach the biosphere.

Mined Repositories in Volcanic Tuff. The DOE’s 2008 license application for the proposed repository at Yucca Mountain in Nevada, USA [12, 16] provides the only example of disposal in volcanic tuff. The concept calls for emplacement of both spent fuel and high level radioactive waste (in the form of borosilicate glass) in corrosion resistant waste packages placed end-to-end in mined tunnels. The proposed facility is 200 to 300 meters below the land surface and, because of topographic relief and the aridity of the surrounding region, is also more than 200 meters above the water table in an unsaturated and oxidizing environment. Primary barriers providing isolation include the low volume of water flow through the unsaturated rock and the long-life expectancy of the waste package (Alloy-22, composed of nickel with high concentrations of molybdenum and chromium and chosen for its high corrosion resistance in oxidizing environments) and the overlying titanium drip shield that (while intact) will prevent seepage water from contacting the waste package surface. When waste packages fail, radionuclide sorption on corrosion products and mineral phases along the transport pathway will reduce the magnitude of releases that may reach the biosphere.

Mined Repositories in Argillite. Published analyses from the Belgian, Swiss, and French programs [8, 9, 17, 18] provide examples of mined repositories in clay-rich rocks. Specific details of rock properties and facility design differ among the three examples, but in each case the chemically reducing conditions in the repository and the lack of advective flow in the low permeability host rock contribute to the long term isolation of the waste. When waste packages fail, mobility of radionuclides will be limited by precipitation in the reducing groundwater, sorption on clay minerals in the host rock, and the slow rate of diffusive transport through the host rock.

Mined Repositories in Salt. The US is currently disposing of intermediate-level transuranic waste in bedded salt at the Waste Isolation Pilot Plant (WIPP) [2], and Germany has both disposed of intermediate and low-level radioactive wastes in domal salt at Morsleben and investigated possible disposal of spent fuel and high level radioactive waste in domal salt at Gorleben [11]. Details regarding potential release pathways and the amount of brine that might contact the waste differ between bedded and domal settings, but in all cases, isolation in salt relies primarily on the extremely low permeability of intact salt (primarily halite), which precludes advective transport of radionuclides away from the repository. Observations made here are also based in part on analysis in the US of the long term performance of a generic bedded salt repository [10].

Deep Borehole Disposal in Crystalline Basement Rock. No national programs are currently pursuing the deep borehole disposal option (3 to 5 km deep disposal), but multiple investigations over the past twenty years have suggested that it may be a viable option for relatively small volume waste forms with physical dimensions suitable for emplacement in holes drilled from the land surface [13, 14]. Attributes of the concept that contribute to long term isolation of the waste include the anticipated conditions at increasing depths: reducing chemistry; decreasing permeability, including the low frequency of open fractures in crystalline (granitic or metamorphic) rocks below 2 to 3 km; and increasing fluid salinity and density, which counters thermally driven upward flow; as well as the extremely long diffusive transport path through the borehole seal system. Published performance assessment analyses conducted in the US [13, 14] provide the basis for the observations below on how changes to waste form properties may impact deep borehole repository performance.

3. IMPACTS OF CHANGES TO WASTE FORM PROPERTIES ON DISPOSAL SYSTEM PERFORMANCE

Reduction in the radionuclide inventory associated with recovery and re-use of fissile isotopes. Recovery and re-use of fissile radionuclides from spent fuel will directly reduce the inventory of those radionuclides, per kW of electricity generated, in the waste for geologic disposal. However, this inventory change does not necessarily produce a proportional increase in the long term safety of the proposed disposal systems. In the absence of disruptions that directly expose waste to the biosphere (such as human intrusion), estimates of the long term performance of disposal systems are dominated by the most mobile radionuclides, rather than those that contribute the most to total radioactivity. As shown in Figure 1, the radioactivity of typical spent nuclear fuel will be dominated for most of the next several hundred thousand years by isotopes of Pu. However, Pu and other actinide elements have limited mobility in chemically reducing environments, and published safety assessments for most disposal concepts show essentially zero direct contribution to risk (in terms of estimated dose) from Pu, U, and other radionuclides proposed to be removed. Risk in these concepts comes instead from long-lived fission and activation products, specifically I-129 and to a lesser extent Se-79 and Cl-36, that are mobile in essentially all geochemical environments.

In contrast, the proposed repositories at Forsmark [6] and Yucca Mountain [12] provide exceptions to the observation that risk is dominated by I-129. Forsmark has reducing geochemical conditions and strong sorption of actinides on the bentonite buffer, and shows no direct contribution to dose from isotopes of Pu or U. The relatively short-lived Ra-226 ($t_{1/2}$ = 1600 years), however, shows up as the primary contributor to dose, exceeding the contribution from I-129 by roughly a factor of 5 [6]. This result is consistent with the chemical mobility of Ra in reducing environments and its continuous ingrowth from immobile Th-230, coupled with site-specific models that allow for relatively rapid transport from the repository to the biosphere in fractures that directly intersect the waste emplacement regions. Comparable Ra-226 releases are not observed in other disposal concepts in reducing environments for which long transport times allow substantial decay of Ra-226 (including in crystalline rock where fractures do not directly intersect the waste emplacement region [7]).

Dose estimates for Yucca Mountain, shown in Fig. 2, show dominant contributions from Pu-238 and Pu-242 throughout the million-year regulatory period, consistent with the relatively higher mobility of Pu (and other actinides) in the unsaturated oxidizing repository environment and oxidizing groundwater transport pathways. Other significant contributions at one million years come from Np-237; Ra-226, which in this case is generated by decay of mobile U-234 and Th-230 throughout the transport pathway; and I-129, which because of its high mobility contributes approximately 1/10th of the total dose at one million years despite contributing less than 1/100th of the radioactivity inventory available for transport at one million years (Fig. 1).

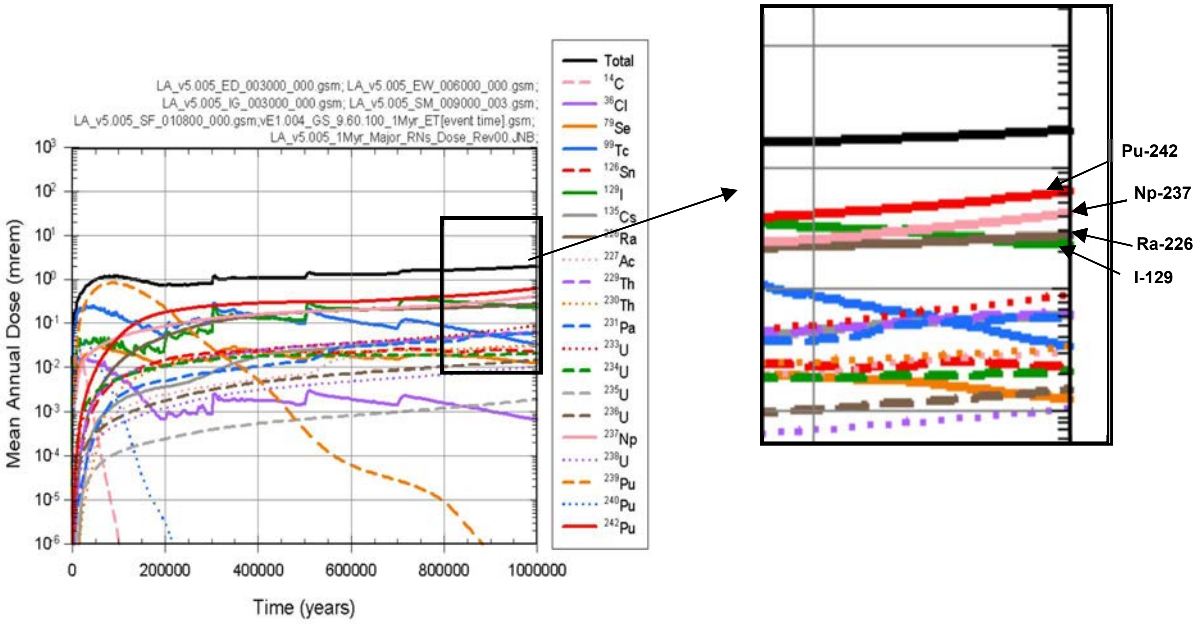

FIG. 2. Estimated mean contributions from individual radionuclides to total mean annual dose resulting from the disposal of spent nuclear fuel and high level glass waste in tuff (from [19], adapted from [12], Figure 2.4-20b).

In summary, radionuclide inventory reductions due to recovery and reuse of fissile isotopes of Pu and U can reasonably be expected to have no effect on estimates of the long term performance of mined repositories in argillite or salt, or in crystalline rock concepts, mined or drilled, in which transmissive fractures do not directly intersect the waste emplacement region. In each case, reducing chemical conditions immobilize actinides and long transport times prevent all but long-lived mobile species, dominated by I-129, from reaching the biosphere in significant quantities. For mined repositories in granitic or crystalline rock where relatively rapid transport to the biosphere may occur in fractures (e.g. Forsmark), removing the actinides that decay to create mobile Ra-226 would have a potential to reduce the estimated total dose by perhaps a factor of 5, at which point I-129 would become the primary dose contributor. Similarly, removing all actinides from the inventory of a repository in oxidizing conditions (i.e. Yucca Mountain) could decrease the estimated total dose by perhaps a factor of ten before I-129 becomes the primary contributor. It should be noted, however, that the currently estimated doses for both Forsmark [6] and Yucca Mountain [12], which include these fissile isotopes, are well below regulatory limits such that there is no reason to suggest they would need to be reduced further to meet safety requirements.

Reduction in the radionuclide inventory associated with additional partitioning and transmutation of radioisotopes remaining after recovery and re-use of fissile isotopes. Nuclear fuel cycles have been proposed that will also remove minor actinides (specifically, Am, Np, and Cm) from the spent fuel in addition to U and Pu. As shown in the discussion above, and noted previously by multiple researchers [20, 23], reductions in the inventory of radionuclides of Am, Cm, and Np will have no perceptible effects on the estimates of total dose from disposal concepts that include reducing conditions and long transport times between the repository and the biosphere. Estimated doses from disposal concepts analogous to Yucca Mountain could be reduced by a small amount if Np-237 and its decay parent Am-241 were removed from the inventory before disposal, but, as noted in the previous section, the effect is limited to roughly a factor of ten before I-129 becomes the dominant contributor. In addition, various approaches have been proposed to transmute I-129 and other fission products into stable isotopes [24, 25]. Transmutation of fission products would clearly impact dose estimates from repositories if they could be implemented cost-effectively at industrial scales, but at this time there is little evidence to suggest that these techniques are realistic options for the future.

Reduction in the volume of waste and thermal power associated with recovery and re-use of fissile isotopes. Waste volume and thermal power are addressed together because, all other factors being held constant, the two properties are inversely correlated. Reductions in volume, unless they are accompanied by the separation and removal of heat-generating radionuclides, increase the thermal power per unit volume of waste. Decreasing waste volume has the potential to decrease the size of the repository and therefore decrease disposal costs, but increases in thermal power of the waste could counter that effect by requiring greater spacing between waste packages to meet repository design temperature constraints. Removal of heat-generating fission products and minor actinides from the waste stream has the potential to reduce waste volume without increasing thermal power, but there are multiple approaches to keep peak post-closure repository temperatures below a specified value that do not include partitioning and transmuting heat-generating radionuclides. For example, waste can be aged before disposal, the repository can be ventilated after waste is emplaced, waste package size can be decreased, and waste package spacing can be increased. Each of these approaches has been proposed in one form or another in published repository design concepts, and thermal constraints do not appear to limit implementation of any of the major disposal concepts for waste forms with a broad range of thermal output.

Although disposing of a wide range of thermal power waste is feasible, separating heat-generating radionuclides can result in substantial reductions in the required total excavated disposal volume. Modeling studies [20] have evaluated variations in the thermal power of the waste and the spacing of waste packages while holding all other aspects of repository design and operations constant. As discussed in [19], the results suggest that the waste from a full-recycle fuel cycle (i.e., including separation of minor actinides) can meet the same temperature constraints for clay and granite repository concepts using only 30% to 40% of the disposal gallery length needed for disposing of waste from an equivalent electric-power-generating open fuel cycle [20]. Doubling the aging time (to 100 years) for waste in which short-lived fission products are the dominant heat sources leads to further reduction in a hypothetical clay repository to approximately 8% of the original disposal volume [20].

For all disposal concepts, waste volume and thermal power considerations are probably best thought of as topics to be addressed through engineering design and cost optimization evaluations, and not as fundamental safety issues for disposal. Estimates of long term dose from published safety assessments [6, 7, 8, 9, 12] meet existing regulatory requirements for waste forms from an open fuel cycle. Alternative waste forms may allow more efficient use of repository space or provide suitable geometries for small-diameter cylindrical waste packages for deep borehole disposal.

Increases in the durability of the waste form in the repository environment. Impacts of waste form durability may be evaluated for disposal of various forms of spent fuel without treatment (e.g., conventional light-water reactor uranium oxide spent fuel [12], TRISO particle spent fuel [26]) and vitrified waste from reprocessed spent fuel (e.g., borosilicate glass). In all disposal concepts, radionuclide releases only occur once the waste form begins to degrade, and those releases then depend on the waste form degradation rate. Increasing waste form durability has been proposed as a means for improving overall repository performance [27], but because many other factors affect the timing and magnitude of the radionuclide source-term and radionuclide migration to the biosphere, the impacts of waste-form durability (i.e., lifetime) need to be evaluated in the context of the full disposal system. Other potentially significant factors include: water flux to/through engineered barriers containing the waste form; degradation rates of the engineered barriers; water chemistry contacting engineered barriers and the waste form; and radionuclide transport properties through engineered and natural barriers. Any of these factors may dominate overall performance of the repository if it controls the dominant radionuclides contributing to estimated dose, but in many cases disparate factors contribute to the dominant radionuclides contributing to the estimated dose. Therefore, it is not always clear whether improved performance of an individual aspect will translate into meaningful improvements of overall disposal system performance. For example, increasing the durability of a waste form may have little impact on the magnitude of the estimated peak dose if for example (a) the dominant radionuclide is solubility limited, or (b) the waste form lifetime is still relatively short compared the transport time to the biosphere.

As discussed in [19], the French safety assessment for an argillite disposal system [9] provides an example in which overall performance is relatively insensitive to waste-form lifetime because modeled releases are largely controlled by the slow rate of diffusive transport through the geosphere. For analyses assuming direct disposal of spent fuel, radionuclides were assumed to be released from the fuel over approximately 50 000 years [9]. Possible sensitivity to this assumption was tested by assuming a ten-fold increase in dissolution rate of the fuel matrix (i.e., a one-tenth lifetime) and it was shown that the time and magnitude of peak releases to the biosphere were essentially unchanged because the reference waste-form lifetime of 50 000 years was already significantly shorter than the transport time through the geosphere [9]. Repository performance showed a somewhat greater sensitivity to increases in the degradation rate (decreased lifetime) of high level glass waste because its reference lifetime for the analysis was longer, “on the order of a few hundred thousand years” [9]. The reference lifetime was based on the degradation rate slowing to a residual rate when the surrounding medium becomes saturated with silica. The sensitivity analysis applied an alternative conceptual model in which glass degradation rates were held constant in time at the initial rate, thus diminishing the waste-form lifetime to thousands of years. Results for this alternative model showed an insignificant increase in the peak biosphere release of the dominant radionuclide contributing to dose, I-129, from 8.6×10^{-4} mol/yr to 9.1×10^{-4} mol/yr, but a substantial shift in the time of peak release from 460 000 yr to 250 000 yr [9].

Results from a preliminary safety assessment for the Swedish granite repository proposed at Forsmark [28] show that, for this example, transport from the repository to the biosphere can occur by relatively rapid advective flow in fractures (on the order of thousands of years). In the base case analyses for corrosion failure of waste packages [28], spent fuel fractional dissolution rates in the reducing environment ranged from 10^{-6}/yr to 10^{-8}/yr, corresponding roughly to waste-form lifetimes ranging from 1 000 000 yr to 100 000 000 yr [29]. The sensitivity analyses of the fuel dissolution rate [28] for cases where waste package failure occurs at 500,000 years, indicate that, within the above range of values anticipated for reducing conditions, estimated dose to an exposed individual in the biosphere varies essentially linearly with the dissolution rate. This sensitivity to longer waste form lifetimes relative to the transport times is consistent with the sensitivity shown in the French repository above for longer glass lifetime but is more pronounced in the Forsmark example because the waste form lifetime is orders of magnitude longer than the transport time. However, for the Forsmark sensitivity analyses using significantly higher dissolution rates (i.e., 10^{-5}/yr and higher – waste form lifetimes roughly 100 000 yr and less), the results are insensitive to waste form lifetime even though the transport time is much shorter. This is because the radionuclide dominantly contributing to the estimated dose, Ra-226, is continuously produced in the waste form from decay of Th-230 (which in turn is produced by decay of U-234). The rate of Ra-226 ingrowth presumably determines the availability of Ra-226 for transport at higher fuel dissolution rates, causing the overall dose to be insensitive to increases in fuel dissolution rate above 10^{-5}/yr.

Other disposal concepts show behavior comparable to that explained in detail for French and Swedish concepts. In summary, waste form durability becomes an important contributor to overall repository performance for disposal concepts where transport time to the biosphere can be relatively short compared to the regulatory period. In concepts where transport from the repository to the biosphere is dominated by slow diffusion that occurs over durations that are substantial fractions of the regulatory period, as in argillite, salt, unfractured crystalline rocks, and deep boreholes, changes in the degradation rate of the waste form may have relatively little impact on the magnitude of the estimated peak dose.

4. CONCLUSION

Insights from published safety assessments for disposal of spent nuclear fuel and high level radioactive waste suggest that modifications to waste forms from potential advanced fuel cycles are not essential for demonstrating safe long term performance of repositories. Modifications that reduce the thermal power of the waste or that reduce waste volume without increasing thermal loading have potential to allow more efficient use of underground mined repository galleries, and potentially also offer pathways to developing waste forms that would fit within deep borehole disposal systems. Changes in the radionuclide inventory of waste forms from the potential recovery and reuse of fissile material contained in spent fuel are unlikely to have a significant impact on the estimates of long term performance for most disposal concepts (in the absence of disruptions that expose the waste directly to the biosphere such as human intrusion) because of the relatively higher mobility of the long-lived fission product I-129 in most disposal system environments. Waste form modifications for durability have the potential to improve estimated peak dose performance of repositories only if the modified waste-form lifetime becomes relatively long compared to the geosphere transport time, and/or approaches the period of performance (e.g., on the order of hundreds of thousands of years). Relatively smaller improvements in waste-form lifetime (e.g., on the order of thousands or tens of thousands of years) may simply delay the time of the estimated peak release to the biosphere.

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DEVELOPMENT OF THE MULTINATIONAL REPOSITORY CONCEPT: EXPLORING ALTERNATIVE APPROACHES TO FINANCING A MULTINATIONAL REPOSITORY

T. ŽAGAR
Head of Planning and Control, GEN energija d.o.o
IFNEC RNFSWG Co-Chair
Slovenia
tomaz.zagar@gen-energija.si

S. TYSON
Office of International Nuclear Energy, Office of Nuclear Energy, U.S. Department of Energy
IFNEC RNFSWG Co-Chair
United State of America
Sean.tyson@nuclear.energy.gov

R. MUSSLER
Nuclear waste policy consultant
mussler.robert@gmail.com

Abstract

The amounts of waste generated in the nuclear power lifecycle is small compared to other power generation options and normalized to power produced [1]. In particular, because of the enormous energy density in uranium, nuclear power plants produce much smaller quantities of waste than fossil plants. Although there are several back–end management options that result in different waste forms for countries generating spent fuel and high level radioactive waste, a geologic disposal capability is required.

High level radioactive waste (HLW) and/or spent fuel (SF) need technologically advanced treatment and management procedures from interim storage to final disposal. To prevent any negative impact on the environment or and human health, HLW and SF must be adequately isolated. Disposal in a deep geological repository (DGR) is internationally recognized as the most technologically developed and safest approach to isolating these wastes from the biosphere. Development of a DGR involves high fixed costs that carry an associated economy of scale. A DGR with a capacity of 10 000 metric tonnes can cost little more than one to dispose of 5000. This means that smaller nuclear programs could benefit greatly from the opportunity to participate in a Multinational Repository (MNR).

The MNR concept provides a shared solution to the challenges of SF and HLW disposal. The concept involves a service provider country developing a geologic repository and accepting SF from several customer countries. Although financing is an issue shared by all repository projects, a MNR project presents a unique case regarding issues associated with the sources of funds, timing of revenues and expenditures, and risk allocation. Different international organizations are approaching this issue from diverse aspects. Recent developments regarding the identification of financing approaches for an MNR have been observed among different fora and will be presented in the paper. These activities include actions of different intergovernmental and international organizations (i.e. IAEA, OECD, WNA [2]), however this paper will focus mainly on results of recent work done by the International Framework for Nuclear Energy Cooperation's (IFNEC) Reliable Nuclear Fuel Services Working Group [3].

1. INTRODUCTION

For the past decade the International Framework for Nuclear Energy Cooperation (IFNEC) has worked to advance the Multinational Repository (MNR) Concept. The concept involves countries that share the challenge of disposing of spent fuel or high level radioactive waste working together toward shared solutions, and has been discussed and developed in a number of IAEA publications going as far back as 1998 [4, 5, 6, 7].

Countries that have programs to develop a national disposal capability while also pursuing opportunities to work with other countries on the MNR concept are following what is referred to as the Dual Track Approach. This approach was perhaps first described in detail in a European Commission sponsored project called SAPIERR II as reported in its 2008 report [8] and was further developed in a report developed by IFNEC in 2016 [9].

One of the key challenges associated with the development of an MNR is financing. As part of the IFNEC work on the MNR concept, a workshop was held in Paris in December 2018 [3], to begin a dialogue on the various approaches that might be used to finance an MNR. This paper summarizes the outcomes of that workshop.

There has been very little work in the past on this topic and the workshop served to begin a discussion that should continue in the future as the MNR concept is further developed and individual country interests in shared solutions increase.

2. WHY AN MNR?

Many countries currently have small nuclear power programs that generate relatively small amounts of spent fuel and/or high- level waste. The number of countries adding nuclear power generation to their energy mix is expected to increase over time, and this will likewise increase the number of nuclear power programs generating relatively small amounts of spent fuel.

There are over 250 000 metric tons of spent fuel in temporarily storage in thirty-three countries worldwide. This number is an estimate given in the latest IAEA publication [10]) from 2018. It takes the total from 2013 367 000 metric tons of spent fuel and subtracts the spent fuel that has been reprocessed (120 000). Almost none of these countries have a clear path to final disposal of this fuel or of the wastes that could arise from its reprocessing. These wastes require technologically advanced treatment and management procedures from storage to final disposal.

The long timescales over which some waste remains radioactive has led to the idea of deep disposal in underground repositories in stable geological formations. Isolation is provided by a combination of engineered and natural barriers (rock, salt, clay) and no obligation to actively maintain the facility is passed on to future generations. Deep geological repository (DGR) disposal is the preferred option for nuclear waste management in several countries including Argentina, Australia, Belgium, Canada, Czech Republic, Finland, France, Japan, the Netherlands, Republic of Korea, Russia, Spain, Sweden, Switzerland, the UK, and the USA [11].

Development of a DGR involves high fixed costs that possess an associated economy of scale. A DGR to dispose of 10 000 metric tons can cost a little more than one to dispose of 5000. This means that smaller nuclear programs would benefit greatly from the opportunity to participate in a project where many countries dispose of their wastes in a single DGR. Accordingly, the MNR concept involves a service provider country developing a geologic repository and accepting spent fuel and high level radioactive waste from several customer countries.

3. CHARACTERISTICS OF A DGR PROJECT

3.1. Phases and spending profiles.

There are four basic phases for the development of a DGR project given below. Some generally applicable durations and spending profiles based on reviewing available information from national DGR programs were estimated and are given here:

- Siting and Licensing – 20 years – 15% of total costs;
- Construction – 15 years – 35% of total costs;
- Operations – 40 years – 45% of total costs;
- Decommissioning, closure, long term monitoring – 75 years – 5% of total costs.

Programs reviewed included France, US, Finland, Sweden, and others.

The nominal 35-year period from project initiation to commencing operations is well beyond that of most construction projects. Although the initial siting and licensing phases do not require large upfront investments, as much as 50% of the total costs are incurred before disposal operations can begin.

Note that these time periods are conservative (based on expectations for a project done today) and present significant financing challenges. If assumptions are added that include completed national experience in developing a DGR, it is possible that the time periods could be reduced to perhaps 13-15 years for Siting and Licensing and 10 years for Construction. Under these assumptions perhaps the lower costs of Siting and Licensing could occur without significant financing, and the financing that would be required for Construction would be for a 10-year period, closer to existing experience in financing project before revenues begin.

3.2. Per Unit disposal costs

Guidance for cost estimation has been published by the NEA, EDRAM, and the IAEA, and some nations have formal guidance on costing major national infrastructure projects extending over long periods.

A DGR project will have both fixed and variable costs that described as:

(a) Fixed
- Site selection and permitting;
- Surface handling facilities;
- Transport infrastructure;
- Access shafts/tunnels;
- Access closure and sealing;
- Environmental monitoring.

(b) Variable
- Emplacement tunnels, vaults, boreholes;
- Disposal operations;
- Encapsulation of SF/HLW.

Estimates for the costs of disposal are around $1 million USD per metric ton of spent fuel. Because of the significant fixed costs, the costs will be lower per unit for large volume repositories and higher per unit for small repositories. The understanding of disposal costs will continue to be projections until there is actual experience with an operational project.

4. MNR FINANCING

Although financing is an issue shared by all repository projects, as a multinational project an MNR presents a unique case with issues such as the sources of funds, timing of revenues and expenditures, and the allocations of risk. The IFNEC workshop on approaches to financing an MNR was intended to serve as a starting point for fostering robust discussions that would identify and develop those issues [12].

In organizing the workshop IFNEC asked a group of international experts on financing and nuclear project development to propose their own creative approach to financing an MNR. The approaches were presented in some detail. The following are brief summaries of each. Note that these approaches are intended to be conceptual, hopefully encouraging further creative thought and discussion. Might you have a better idea?

4.1. Approach 1: It is clearly challenging to finance one MNR…but may be easier to finance several.

This approach assumes a consortium of countries in different regions of the world interested in developing an MNR. The first MNR (MNR-1) will have the largest risk in terms of siting, licensing and construction, however all participating governments would share the upfront risk for siting, licensing and construction. Private funding will come in during the commercial operation. Based on a harmonized approach and replication to the extent possible, risk for MNR-2, MNR-3, MNR-4, etc. should decrease. Siting, licensing and construction times should also decrease.

4.2. Approach 2: Two Approaches: government lead with and without customer investment

This presentation identified two approaches to financing. The first approach (Option 1) involves the government developing the project and providing initial financing from development through initial operation (waste emplacement), at which point an exit strategy (in part) could be utilized. The second approach (Option 2) focuses on the early financial participation of the Customers through the purchase of shares in the repository project, with finances managed in an arms-length fund. In the case of Option 1, the government leads the overall effort, with its role decreasing over time. With Option 2, the effort is co-led by the government and one or more customers that take membership interests in the project.

Both Options assume the following:

(a) The government will need to provide overall leadership with an underpinning of public and political support, legal & regulatory regimes, and the necessary supporting infrastructure;

(b) Initial participation, while supported by commercial commitments, will rely on government-to-government relationships to establish participation from customer countries, with contractual commitments that are backstopped by sovereign guarantees.

4.3. Approach 3: Sell shares in the repository project with return on investment coming from fees collected during operation

This approach to financing posits a country sponsoring the development of a geologic repository project through a state-owned agency that would be empowered to enter into multilateral agreements with other countries selling equity shares in the project. Shares would be sold in a venture capital style model, with funding rounds reflecting the project's current status. Parties to the project would appoint a trustee to manage the funds to ensure transparency.

4.4. Approach 4: Financing with a staged interim storage/repository approach

A staged repository consists of an initial phase of developing and operating a spent fuel storage facility (dry storage) with a portion of revenue allocated to development of a co-located repository. This approach would facilitate commercial investor involvement in a step-wise fashion and build credibility and experience for nuclear fuel management by phasing out "by and for governments only" and replacing with commercial investment based on optimized cost and knowledge management. It includes selling shares in the repository project with return on investment coming from fees collected during operation.

3. CONCLUSION

As a matter of policy, countries that generate spent nuclear fuel set aside funds to support disposal. Those funds could be used, depending on national authorities, for developing in-country disposal capability, or purchasing an international disposal service. Funding for an MNR exists today. The financial challenge lies in identifying the financing arrangements for developing the project that are attractive for all stakeholders: governments, the service provider, the customer, investors, etc.

The Workshop was an initial effort to identify non-traditional financing approaches for a unique case, i.e., financing the construction and operation of an MNR. It is interesting that even though independently developed, there are common themes in the approaches presented. Each approach may present opportunities for further consideration and analysis.

There is considerable international interest in the MNR concept. IFNEC is pleased to have initiated what is hoped will be the first of many further discussions on this topic. We suggest that further discussions addressing approaches to financing an MNR could be the key to unlocking the first MNR project.

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Paper ID#130
STAKEHOLDER COMMUNICATIONS AND ENGAGEMENT IN THE SITE SELECTION PROCESS FOR CANADA'S DEEP GEOLOGICAL REPOSITORY FOR USED NUCLEAR FUEL

L. FRIZZELL
Vice President, Stakeholder Relations
Nuclear Waste Management Organization
Toronto, Ontario
Email: lfrizzell@nwmo.ca

Abstract

Canada's Nuclear Waste Management Organization (NWMO) is leading a site selection process for an informed and willing host community with a suitable site for a deep geological repository for used nuclear fuel, as well as an associated Centre of Expertise. The process was initiated in 2010 and is expected to culminate with identification of a preferred site around 2023. It is a community-driven process designed to address a broad range of social, economic, cultural and technical factors identified through dialogue with Canadians and Indigenous peoples. The process involves a step-wise approach with clear decision points, and increasingly intensive stakeholder engagement and technical study. Consistent with the NWMO's commitment to involving people in its work, the siting process is being implemented in an open, transparent and inclusive manner through a growing set of engagement and communications programs. These programs are frequently shaped by the very stakeholders they aim to engage, and seek to: build awareness, understanding and support among key audiences; work collaboratively to identify potential repository sites that are socially acceptable and respectful of social and cultural values; and explore potential to build supportive partnerships to implement the project while enhancing well-being and building resilience of communities. This paper provides an overview of the site selection process, with a focus on approaches used to engage and communicate across a wide range of audiences and platforms to achieve the goals described above. It explores the types of programs and activities used to engage citizens in developing Canada's plan and the site selection process, and in implementing the project collaboratively with municipal and Indigenous communities. It also discusses how the NWMO is expanding and adapting the activities, tools and platforms it uses to increase visibility and understanding of its work among key audiences in preparation for site selection.

1. INTRODUCTION

The Nuclear Waste Management Organization (NWMO) is leading Canada's plan for the safe, long term management of its used nuclear fuel. The approach it is implementing, Adaptive Phased Management (APM), requires centralized containment and isolation of the country's used nuclear fuel in a deep geological repository [1].

The plan is built on a set of principles that reflects the values and priorities Canadians identified on this issue during a three-year, nation-wide dialogue that took place between 2002 and 2005. The repository must be located in a suitable crystalline or sedimentary rock formation, in an area with informed and willing hosts. That means people in vicinity of the site that is ultimately selected must be aware of the project, understand what it would mean to implement it in the area, and support having it located there. The approach is also designed to ensure safety, security and protection of people and the environment are priorities, and to be consistent with international standards and best practices. Given the long term nature of the project, the plan is also adaptive, with adjustments as needed to incorporate new knowledge or societal priorities.

After the plan was selected by the federal government in 2007, the NWMO conducted additional engagement with Canadians to develop the decision-making framework for selecting a site for the project. The site selection process is a community-driven process designed to address a broad range of social, economic, cultural and technical factors as identified through dialogue with Canadians and Indigenous peoples [2].

This site selection process was initiated in 2010 following two-years of development that took into account the suggestions and advice received over the course of public dialogue [3]. The process is voluntary in nature, and twenty-two communities expressed interest in learning about the project and exploring their potential to host a site. The site evaluation process involves a step-wise approach with extensive public engagement and clear decision points.

Over the years, stakeholder engagement and technical study have intensified as the NWMO narrows its focus on fewer candidate sites, while continuing to work with communities to evaluate suitability against a number of social and technical evaluation factors.

Today, following a gradual narrowing down process, five of those original 22 communities remain in consideration. At the same time, engagement and communications has expanded to include a constellation of communities in the vicinity of each siting area, recognizing that the approximately $24-billion (CAN$) project is large enough to impact a region, and that it will only successfully move ahead through partnership with both Indigenous and municipal communities. The process is expected to culminate with identification of a preferred site around 2023.

To identify a preferred site, the NWMO has three main areas of focus: 1) building confidence that the site ultimately selected will be safe; 2) building confidence that a safe, secure and socially acceptable transportation plan to move the used fuel to the repository can be developed; and 3) ensuring willingness to proceed and established partnerships in potential host communities. This paper discusses communications and engagement activities used to achieve these goals, with an emphasis on achieving willingness and partnership.

Consistent with the NWMO's commitment to involving people in its work, the siting process is being implemented in an open, transparent and inclusive manner through a growing set of engagement and communications programs, which are shaped in part by the very stakeholders they aim to engage.

These programs seek to:

- build awareness, understanding and support among key stakeholders;
- work collaboratively to identify potential repository sites that are socially acceptable and respectful of social and cultural values; and
- explore potential to build supportive partnerships to implement the project while enhancing well-being and building resilience of communities.

Since the siting process began, the NWMO has focused much of its engagement efforts on building awareness and understanding of the project in local siting areas. As the prospect of identifying a preferred site draws nearer, the NWMO is also increasingly focused on broadening the reach and frequency of communication and engagement activities to increase awareness among interested citizens beyond siting regions. With this goal in mind, the organization is increasing proactive outreach with media, social media communities, industry, governments and other groups and individuals with an interest.

2. ENGAGEMENT AREAS OF FOCUS

Engagement and communications activities at the NWMO are designed with strategic outcomes in mind. Since the project can only succeed through extensive public alignment and engagement, the organization has expended significant time, effort and expertise on developing strong communication and engagement programs aimed at contributing to conditions necessary for achieving strategic goals. Currently, three of these priorities see the organization actively engaging to:

- Help define and establish the types of partnership that will be necessary to successfully identify a preferred site;
- Shape the technical studies that will be required to demonstrate confidence in the project's safety; and
- Respectfully interweave indigenous knowledge into its activities, processes and decisions.

2.1. Working toward partnership

In order to select a site that can be socially acceptable, the NWMO needs to be confident it can develop strong, aligned partnerships with municipalities, First Nation and Metis communities in the area. Alignment of the project with the values, priorities and objectives of surrounding communities and Indigenous peoples, together with their level of interest in learning, is a critical consideration in assessing the suitability of any particular site.

The APM project will only proceed with the involvement of the interested community, Indigenous people (i.e. First Nation and Métis communities) in the area, and surrounding municipalities all working together to implement it. Through its work with communities, the organization has come to understand that such partnerships need to be underpinned by a willingness to proceed among people living in these communities, an understanding of how the project will enhance local well-being, and eventually, draft agreements that outline a common understanding of how the project will be implemented.

Given that there is no clear template for how to accomplish such partnerships, the NWMO has worked with communities to develop a roadmap (Fig. 1). The first step is agreeing on common values and principles to guide partnership discussions. Through a series of public workshops, this step was completed in all of the municipalities remaining in the site selection process. The values and principles developed were subsequently passed by each municipal council, effectively paving the way for more detailed discussions.

The next step in working toward partnership is collaboratively developing a project vision. This involves developing a common understanding of what the project could look like and how it could be implemented in each area. The NWMO also plans to work with communities to understand the nature of partnerships that need to be developed, understanding with whom it needs to work, at what level, in what combination and on what timelines. Further, the organization will need to identify and deliver investments that help communities build the capacity they will need to take on a project of this scale in a way that advances local well-being and supports achievement of their own community vision. Finally, the organization anticipates working together with communities to develop aligned partnership agreements, using a schedule developed and agreed upon together.

FIG. 1. The NWMO has worked with communities to develop a roadmap to partnership.

2.2. Shaping studies and plans

Since initiating the site selection process, the NWMO has actively worked together with community leaders and residents to collaboratively shape both engagement activities and technical studies. This approach has provided significant value both to the organization and communities as they work together to explore the potential in each area for hosting the project. Local input has influenced timing, approach and method of communication related to a number of studies.

For example, the NWMO has begun working with people in each area to identify specific sites that could potentially host a repository, taking into account both safety requirements and perspectives of those in the area. In one siting area, borehole studies were successfully initiated in 2017. This was a significant development as it represented the first subsurface studies on a specific site where a repository could eventually be located. The site itself was selected with extensive input from local municipal and First Nation residents about where exactly the project had the potential to be socially acceptable. Aspects ranging from current land uses to cultural significance were all considered. Studies on the site are continuing, with subsequent boreholes planned in 2019 [4].

In addition to engaging on specific local activities, the NWMO seeks broad public input on its overall project implementation plans. These strategic plans outline activities the organization expects to undertake over the next five years and are updated on an annual basis. Public input is sought each year through the NWMO web site, mail, social media, and community engagement activities. The insights received are acknowledged and incorporated into the subsequent year's plan along with updates related to the ongoing implementation of APM.

2.3. Interweaving Indigenous knowledge

As part of the NWMO's promise to work in partnership with First Nation and Métis communities, it has committed to interweaving Indigenous knowledge throughout its plans, activities and decision-making processes [5]. This commitment recognizes that Indigenous peoples have a special relationship with the natural environment, and unique stewardship responsibilities that are part of this relationship.

Indigenous knowledge emphasizes the interrelationships between all components of the environment. It is a complex and sophisticated system of knowledge drawing on millennia of wisdom and experience. It constantly grows and expands with the experience of new generations. The knowledge that comes from this relationship with the land brings special understanding to the broad range of factors, providing a source of wisdom to field studies, social assessments, assessing benefits and effects to be managed.

The NWMO actively works with a Council of Elders and Youth, First Nation and Metis communities, and a range of Indigenous organizations. With their guidance, the NWMO is working to ensure it respectfully applies traditional knowledge in activities and decision making related to both technical and community engagement aspects of the site selection process.

Indigenous knowledge has directly affected the NWMO's work, and in particular the way it engages and communicates. In one recent example, Indigenous communities expressed a need to understand more about how the project will protect water. In response, the NWMO brought together scientists and Indigenous people to create a presentation about the "Journey of Water," exploring how water behaves on the surface and deep underground. The presentation was drafted and refined with feedback from the Council of Elders and Youth, an Indigenous community and a group of Indigenous women, recognizing that in Indigenous culture women in in particular hold a special relationship with water. Once it was finalized and delivered to a range of interested communities and groups, the presentation received remarkably positive feedback from both Indigenous and non-Indigenous audiences. Subsequent presentations are now also being developed using a similar approach.

3. COMMUNICATION TOOLS AND ACTIVITIES

The process of engagement and communication has become increasingly intensive as the project has advanced. Consistent with the NWMO's commitment to involving people in its work, the siting process is being implemented in an open, transparent and inclusive manner through a growing set of engagement and communications programs. These programs and activities are frequently shaped by input and guidance from the audiences they aim to engage. The responsive nature of this approach has likely contributed to the fact that participation among community leaders and residents has remained strong, in spite of long time frames and growing intensity.

The NWMO holds transparency as one of its core values, with a commitment to be open and transparent in processes, communications and decision-making. To demonstrate this commitment, it has developed and published a transparency policy [6]. In practical terms, the NWMO demonstrates this commitment by actively communicating about and documenting new developments and next steps as it implements Canada's plan, outcomes of technical studies, and even input it receives from the public (with permission).

A wide variety of approaches are used simultaneously to seek input, report progress, and address questions and concerns. The following examples do not represent an exhaustive list, although they do demonstrate the wide range of platforms through which people can learn more, share their thoughts, and participate in the process.

3.1. Resourcing to Support Learning

The NWMO provides support to communities in the form of funding programs and expertise to cover the costs associated with learning about the project [7]. It has committed that no community should be 'out of pocket' for learning about the project and exploring their potential to host it. Examples of typical costs include those associated with staffing, studies, conference attendance, meetings of community liaison committees and a wide range of engagement and learning activities. As the project advances and work associated with the project intensifies, investments in capacity building have become particularly important. These programs enhance communities' capacity both to participate in the process and to begin to prepare to host the site if their region is ultimately selected.

In addition to funding, the NWMO also employs relationship managers to lead local engagement activities with municipal and Indigenous communities in each region. More broadly, the organization's stakeholder relations and engagement teams actively seek to communicate about its work with audiences beyond siting communities, including industry, government, media, social media communities, NGOs and other groups and individuals with an interest in the project. Just like activities within potential siting communities, the intensity and frequency of this broader public outreach has also intensified as the NWMO approaches site selection and seeks to raise awareness and understanding about its work.

3.2. Engaging with Community Liaison Committees (CLCs)

Each municipality remaining in the site selection process has assembled a CLC comprised of volunteers committed to facilitating learning in the area. Meetings are advertised and open to the public, and the NWMO participates regularly by sharing updates, bringing subject matter experts on a range of topics, and answering questions about various aspects of the project.

These committees provide an important source of long term continuity and provide a forum where community questions and concerns can be heard and responded to. They also provide residents with information about a variety of viewpoints and specialized knowledge through programs that bring guest speakers to the community.

CLC members get actively involved in activities such as hosting and co-hosting engagement events, directing investments in education and skills development and providing feedback about how the NWMO can best communicate with the public in a manner that is clear, transparent and accessible to those with a wide range of knowledge levels and with a diverse set of communication preferences. Over time, some of these committees have begun to shift their role from learning about the project to advocating for it.

3.3. Hosting and participating in events

The NWMO regularly works with local municipalities and Indigenous communities to host public open houses and workshops in communities to engage residents with an interest in learning about the project and providing input into next steps. The NWMO often brings subject specialists and interactive exhibits to these events to help facilitate learning and build understanding about various aspects of the project. The events are promoted through media, and local media are invited and encouraged to report on them. Increasingly, the NWMO promotes and reports on these events through its various social media platforms as well.

In addition to NWMO-hosted events that are specifically focused on Canada's Plan, the organization has made it a practice to participate with information kiosks and staff at local events that draw significant community participation, such as fairs and festivals. These activities help the NWMO expand its local reach, providing opportunities to engage directly with people who may not have been interested or available to attend an open house or public meeting specifically focused on the topic of Canada's plan for used nuclear fuel.

Outside of siting areas, the NWMO regularly organizes tours of interim storage facilities and its own proof test facility. The tours are typically accompanied by an in-depth briefing about Canada's plan and the process for selecting a site. These events allow members of interested stakeholders to see first-hand how used nuclear fuel is currently stored, and the rigorous work underway to build confidence in the safety case for the deep geological repository (Fig. 2).

FIG. 2. This image was taken during a tour of the NWMO's proof test facility. The organization hosts tours regularly, so members of the public can see first-hand the rigorous work underway with respect to Canada's plan for used nuclear fuel.

The NWMO also hosts events and participates in a variety of conferences that bring together key audiences beyond siting communities. These include nuclear industry conferences, municipal conferences, and special interest conferences that feature areas of expertise relevant to the project such as geology, impact assessment and communications. Not only are these conferences a helpful way to learn from others, they provide opportunities for the NWMO to keep others abreast of its work through presentations, information kiosks and networking activities.

In one recent example, the organization hosted a two-day workshop that brought together western scientists and Indigenous knowledge keepers. During the workshop, participants shared information and perspectives on how Indigenous Knowledge and western science can be interwoven into research applications pertaining to the repository and the multi-barrier system that will be used to contain and isolate the used nuclear fuel. Important insights were brought forward from both knowledge systems about topics such as copper, clay and rock – all materials that will be used to ensure isolation of used nuclear fuel in the repository [8].

3.4. Deploying interactive exhibits

The NWMO has learned through extensive public feedback and experience that communicating in ways that are interactive, relatable and tangible are particularly effective at building interest and understanding. To that end, the NWMO has done developed and deployed exhibits that appeal to a wide range of age groups and knowledge levels.

A few examples include:

- Installing interpretive exhibits and regularly hosting tours in its proof-test facility so that the public can see first-hand the work underway to ensure people and the environment will be protected;
- Developing local community offices, known as Learn More Centres, featuring displays, props and literature that bring to life a range of topics related to the project, how it will ensure safety and how people are engaged; and
- Deploying a traveling exhibit that is used to help tell the story of Canada's plan through interactive modules using featuring images, props and plain language text (Fig. 3).

FIG. 3. This image shows people interacting with one of the NWMO's exhibits.

3.5. Engaging digital communities

North Americans are extremely engaged in social media, with 88 per cent of the population actively using social media platforms [9]. Given this context, social media platforms have become important tools for sharing information about corporate activities with a wider audience.

The NWMO is still relatively new to social media, having launched its presence on Facebook in 2017, and Instagram and Twitter in 2018. However, through posting engaging content about the project and the people involved with it, followers and engagement through these platforms has grown steadily.

The NWMO uses social media to bring additional transparency to its work by sharing updates about new developments and activities, answering questions, and addressing common misconceptions. Social media platforms allow the organization to more effectively engage with audiences beyond potential siting communities. At the same time, these platforms also serve as a digital complement to local, in-person engagement efforts within these communities.

The organization regularly analyses what type of content attracts attention from which audiences and the types of questions it receives over these platforms. As a result, it is able to continuously improve its practices to be responsive to public interest. The NWMO strives to use social media to tell its story in a way that directly and positively engages with key audiences. Using a mix of images, videos and plain language copy, content focuses on topics such as questions and answers about technical aspects of the project, profiles of subject experts, and local community investments in organizations that promote well-being in the area.

The NWMO's social media presence is underpinned by a robust web site www.nwmo.ca which functions both as a communications tool and a transparent archive of the organization's work. Features include a library of reports, a database of shareable questions and answers, frequent news stories, and dedicated landing pages outlining completed and upcoming work in siting areas.

3.6. Engaging media

Media outlets provide both an important audience and a platform for reaching other key audiences with an interest in Canada's plan for used nuclear fuel. Different types of media – such as local newspapers and radio stations, specialized industry or science media, regional television stations, as well as provincial and national media – have all reported on the project from different angles and at different points in time. At the same time, reporters in Canada are facing increasing resourcing challenges, which sometimes make it difficult to for them to report on complex stories in a timely and accurate way.

To help ensure complete and accurate reporting on Canada's plan for used nuclear fuel, the NWMO undertakes a number of activities to build relationships and knowledge among media. Examples include:

— Media tours and briefings at nuclear facilities where the waste is currently stored, so journalists can see first-hand the rigor involved in managing used nuclear fuel and ask questions of specialists about how safety and security will be ensured in the future;
— Proactive outreach to journalists about new developments with the project, or specific areas of interest that align with their media outlets' goals or areas of interest;
— Photo and video content that help reporters tell the story in an engaging way, whether their story will be delivered online, in print or via broadcast; and
— Media training of key subject matter experts to equip them to deliver successful interviews with concise, straightforward and consistent key messages.

In addition to earned media coverage generated through reporters, the NWMO is also reaching audiences through paid media such as advertising and paid editorial content. These tools allow the organization to proactively reach key audiences with a higher frequency and specific messages about the project and ways to engage or learn more about it.

3.7. Internal communications and skill-building programs

As the project advances and the NWMO seeks to continuously increase public awareness, it sees communications and engagement as an essential company-wide endeavor. The organization has recognized that all employees are potential ambassadors for the project. It is common for technically focused employees to join the engagement team in communities, so the local public can hear directly from specialists about various aspects of the project. People from across the company have staffed information booths at industry conferences, and employees with a range of skillsets have been trained to deliver tours at the organization's proof-test facility. All employees are encouraged to share ideas for engaging content, to network with others in their field of expertise, and to continue to advance their knowledge and understanding of the project so they are equipped to answer questions.

To that end, the organization has implemented internal communications programs to help keep employees informed of new developments, not only through their managers, but also through activities such as monthly Lunch and Learn lectures that delve into different types of work happening across the organization, chat sessions with the CEO and other members of the executive team, and annual staff conferences that bring together the entire workforce to share information.

The organization has also implemented communications training programs in areas such as presentation skills, media interviews and social media to help employees continue to learn and build experience.

4. CONCLUSION

The NWMO's extensive engagement and communications programs are frequently shaped by the audiences they seek to involve. Canada's long term plan for used nuclear fuel and the site selection process were both based on extensive public dialogues. In turn, the process is being implemented in an open, transparent and inclusive manner that seeks to incorporate advice and guidance from key stakeholders, both local in siting communities as well as Indigenous peoples and the broader Canadian population. Taking cues from the feedback it receives, the NWMO is actively expanding and adapting the activities, tools and platforms it uses to increase visibility and understanding of its work as it prepares for site selection.

The programme has been highly successful to date. When the siting process was launched in 2010, 22 communities stepped forward to get involved. Engagement remains high in those still involved, even though many years have passed. Communities have agreed on the values and principles required to achieve partnership. With guidance from elders, youth and communities the organization has developed an increasingly strong approach to interweaving Indigenous knowledge. Beyond communities, the organization is seeing steady growth and positive public engagement through media, social media and industry and other platforms.

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SAFEGUARDS-BY-DESIGN FOR ENCAPSULATION PLANTS AND GEOLOGICAL REPOSITORIES

K. BAIRD, K. KHRUSTALEV, S. POIRIER, I. TSVETKOV
International Atomic Energy Agency
Vienna, Austria
K.Baird@iaea.org

Abstract

This paper focuses mainly on the safeguards by design and how the IAEA works with Member States to implement safeguards but aspects on the safeguards for encapsulation plants and geological repositories is also addressed. The Safeguards-by-Design concept is introduced which facilities the collection and evaluation of the safeguards-relevant information, from the initial planning of a new nuclear facility through design, construction, operation and decommissioning. The successful case of implementation of the Safeguards-by-Design concept is illustrated in the case of the Onkalo geological repository and encapsulation plant for spent fuel in Finland.

1. INTRODUCTION

Safeguards refers to a set of technical measures applied by the IAEA on nuclear material and activities, through which the Agency seeks to independently verify that nuclear facilities are not misused and nuclear material not diverted from peaceful uses. The Member States accept these measures through the conclusion of safeguards agreements. Through its experts, IAEA conducts extensive verification missions, collecting and evaluating safeguards-relevant information.

For the States with comprehensive safeguards agreements, safeguards is ensured by addressing three generic technical objectives:

— detecting any diversion of declared nuclear material in declared facilities or Locations Outside Facilities;
— detecting any undeclared production or processing of nuclear material in declared facilities or Locations Outside Facilities;
— detecting undeclared nuclear material or activities in the State as a whole.

A set of technical measures are applied by the IAEA on nuclear material and activities, through which the Agency seeks to independently verify:

— that nuclear facilities are not used for undeclared production or processing of nuclear material;
— that nuclear material is not diverted from peaceful uses;
— the absence of undeclared nuclear material and activities.

2. SAFEGUARDS-BY-DESIGN CONCEPT

The safeguards measures are more difficult to implement at facilities which were not designed with this concept, called Safeguards-by-Design (SbD) [1]. This concept evolved in the past years due to the difficulties encountered in the verification process.

As a definition, SbD refers to the integration of features to support IAEA safeguards into the design process for a new nuclear facility, from the initial planning through design, construction, operation and decommissioning. Thus, costly and time-consuming redesign works or retrofits of new nuclear facilities for implementing the IAEA safeguards approach can be avoided, improving in the same time the effectiveness and efficiency of the IAEA safeguards. For achieving Safeguards-by-Design, the IAEA establishes dialogue as soon as the engineering design is mature, with the designers, engineers, vendors, operators, subcontractors, State authorities raising awareness on the safeguards requirements and needs in terms of verification access and equipment and infrastructure.

3. SAFEGUARDS-BY-DESIGN STRUCTURE, APPROACH AND PROCESS

The general safeguards needs are anticipated based upon the current safeguards practices and the IAEA guidance. The aim is to incorporate the safeguards infrastructure measures into the nuclear facility design to accommodate a range of IAEA safeguards approaches. Safeguards approaches rely upon IAEA equipment that the nuclear facility design needs to accommodate and SbD should provide flexibility for different IAEA safeguards approaches over plant design life.

In order to comply with the SbD concept, the State has to identify and submit design information for a new nuclear facility to IAEA Safeguards already in the planning phase of the facility. With the Agency support, a Low Level Liaison Committee for SbD is established with the representation of all stakeholders (the IAEA represented by its safeguards inspectors and the Technical Services - SGTS, State safeguards authorities, facility operators and relevant contractors, State nuclear regulatory authority, and regional authorities, if applicable). The Committee will conduct technical meetings to reach a good understanding on how safeguards can be implemented and what measures have to be applied. The process is iterative and its coordination is defined together with the IAEA aiming the development of the safeguards infrastructure requirements, including the development of new safeguards methods if needed.

The early integration of safeguards equipment into the facility design is not currently a practice to consider. Hence, in the planning phase, a joint review of the 3D model is performed and an agreement on the applicable industrial standards is reached, taking into consideration a continuous adaptation of the safeguards equipment to potential facility design changes. It is also possible to agree on cost sharing arrangements, particularly for joint use and maintenance of equipment. Early discussions on the content and format of the required operator declarations are also conducted in the process. These can be daily declarations, depending on the facility design and plan of its operation and safeguards approach.

4. BENEFITS OF THE SAFEGUARDS-BY-DESIGN

There are certainly several benefits that the Safeguards-by-Design approach bring when considered in the early planning phase of a nuclear facility:

— the chances of successful safeguards requirements, implementation and incorporation into the facility design are substantially increased. Thus, the need to retrofit and license updates for the installation of safeguards instrumentation will be reduced, facilitating a cost-efficient safeguards implementation.
— the operator burden is diminished with optimization of the inspector time in the facility. It is a good practice for the operator to know well in advance what is expected in regard to the safeguards.
— the use of facility design/operator process information for safeguards purposes is enhanced.
— the flexibility for future safeguards equipment installation is improved.
— facilitate the join-use of equipment (operator/IAEA).

5. SAFEGUARDS FOR THE GEOLOGICAL REPOSITORIES AND ENCAPSULATION PLANTS FOR SPENT FUEL

An example of a successful use of Safeguards-by-Design concept is the implementation and development of safeguards for geological repositories and encapsulation plants.

The number of geological repositories under design and in construction is increasing. Many countries now consider spent fuel not a waste anymore but a valuable nuclear material for future use in energy generation. Hence it must be safeguarded according to the IAEA policy. In this context, spent fuel needs to be safeguarded even after its placement in a geological repository.

Safeguarding geological repositories is challenging as reverification is practically impossible for disposed spent fuel. The safeguards measures implemented by IAEA are mainly non-invasive and for the geological repositories the non-invasive containment monitoring technologies are key.

The Onkalo geological repository in Finland is the closest to become operational. It consists of two components:

— the encapsulation plant on the surface, for which the safeguards approach and infrastructure requirements were finalized in 2018.
— the geological repository consisting in a system of tunnels at 450m depth, for which the safeguards approach and infrastructure requirements will be finalized in 2020.

When operational, it is anticipated that the geologic repository will dispose of a canister per week. A number of penetrations through the geological repository which need to be monitored are anticipated:

— the encapsulation plant building (the canister lift shaft and controlled area exhaust stack);
— the vehicle access tunnel entrance;
— the ventilation and hoist buildings (personnel lift and geological repository ventilation system).

The spent fuel is transferred from the nuclear power plants to the encapsulation plant in casks and then it is packed into copper disposal canisters which are inserted at a depth a about 450m in the tunnels of the geological repository. Figure 1 illustrates a schematic representation of the Onkalo geological repository.

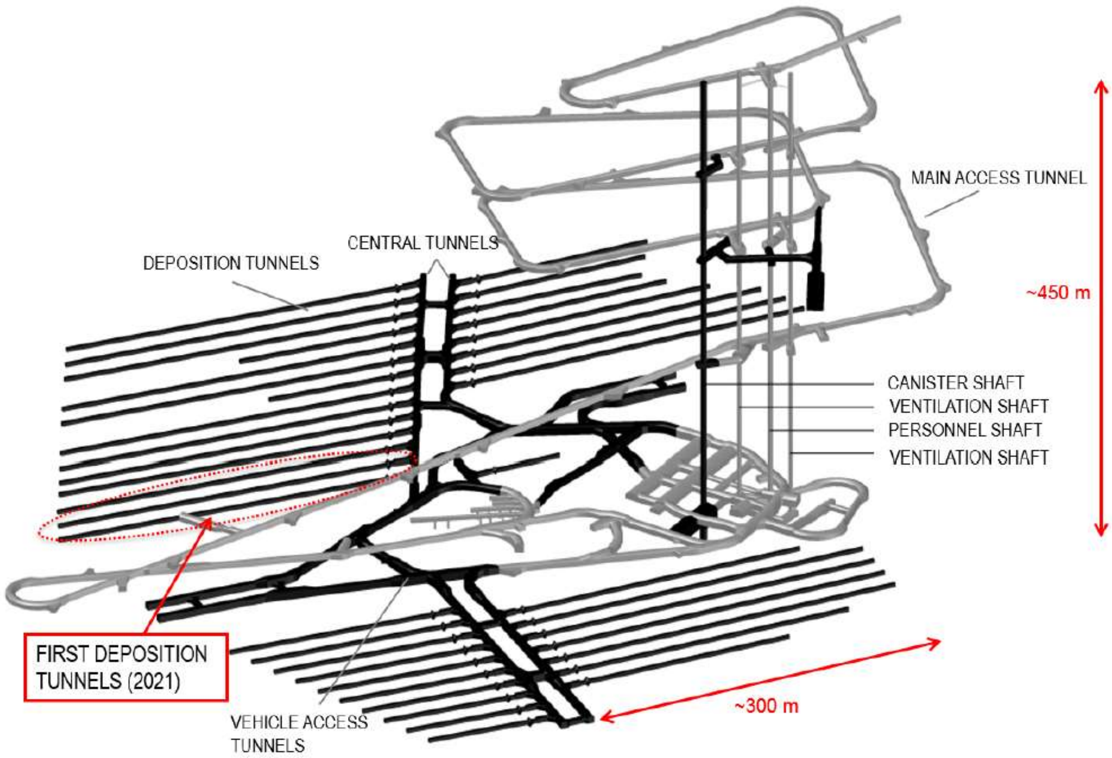

FIG. 1. Schematic representation of the Onkalo geological repository.

For achieving Safeguards-by-Design, the actual implementation of safeguards was preceded by years of relevant dialogue with all stakeholders (the operator – Posiva Oy, the Finnish Nuclear Safeguards Authority – STUK, Euratom and the IAEA) and since 2011, particular study on safeguards for spent fuel at geological repositories and encapsulation plants through SAGOR/ ASTOR group meetings.

Multiple site surveys, as well as design information verification activities (visual observation, 3D laser scanning of the excavated tunnels) were conducted and models based on actual safeguards approaches for the particular encapsulation plant and geological repository were developed.

The discussions on the technical details of the safeguards equipment infrastructure requirements and their integration into the facility design are carried out through the Low Level Liaison Committee.

The majority of the equipment anticipated to be used in the geological repository and encapsulation plant was identified and agreed with the operator to be incorporated in the design. Novel spent fuel verification methods and containment monitoring technologies are being investigated by the IAEA.

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Paper ID#150
DECAY HEAT AND CHARACTERIZATION OF SPENT NUCLEAR FUEL FOR REPOSITORIES AND TRANSPORT

A. SJÖLAND
Swedish Nuclear Fuel and Waste Management Company (SKB)
Stockholm, Sweden
Email: anders.sjoland@skb.se

Abstract

For the disposal, intermediate storage and transport of spent nuclear fuel a number of properties of each fuel assembly must be determined, both for operational and safeguards needs. Important examples of these parameters are decay power, multiplicity, burnup (BU), initial enrichment (IE), cooling time (CT), completeness of fuel assemblies, weight, amount of fissile material and nuclide inventory. This is done through a combination of known fuel history, measurements and codes. In addition, the status of the fuel assemblies is necessary to characterize. Failed or damaged fuels must be identified prior to final disposal in order to treat them appropriately, as are other mechanical and chemical issues that may affect the handling in the system. The uncertainties of these determinations are crucial in the use of the parameters and are judged to be fairly large at present. Particularly the uncertainly of the decay power has a direct relationship to the cost of any repository due to temperature requirements in the systems. These cost savings are potentially very high, in the order of billions of Euros. A thorough understanding of these issues also opens ways to optimize the facilities, for example economically and environmentally. Due to the large amount of fuel assemblies to be measured, high through-put and robustness of the methods and instruments are paramount, as is the capacity to make fast decisions made on the measurement results and codes. The status and future needs of development of instruments, basic fuel data and cross sections, and codes is discussed in the paper, and how this is done in various collaborations world-wide. Potential problems, such as errors in fuel data, uncertainties in basic nuclear data, uncertainty propagation, conflicting methods and results etc., is illustrated and discussed. An international effort to blindly test the capacity to calculate decay power on fuel history, led by SKB and in collaboration with NEA/OECD – with more than 25 participating organizations and groups, using most of the internationally available codes, is described.

1. INTRODUCTION

For the disposal, intermediate storage and transport of spent nuclear fuel a number of properties of each fuel assembly must be known, both for operational and safeguards needs. Important examples of these parameters are decay power, multiplicity, burnup (BU), initial enrichment (IE), cooling time (CT), completeness of fuel assemblies, weight, amount of fissile material and nuclide inventory. These can be determined through a combination of known fuel history, measurements and codes.

In addition, the status of the fuel assemblies is necessary to characterize. Failed or damaged fuels must be identified prior to final disposal in order to treat them appropriately, as are other mechanical, chemical and other integrity issues that may affect the handling in the system. Some of these properties are less direct than the parameters mentioned in the previous paragraph and will be more challenging to clearly describe and to put into requirements.

It is importance to realize that there are several scientific communities that characterize the fuel, in different ways, but often the same fuel property. These are for example fuel ’physics’, fuel ‘chemistry’ and the safeguards verification methods community. These have historically not collaborated to a large extent, although this has somewhat changed the last years. Not least because this type of nuclear research is expensive, substantial improvement in efficiency, accuracy, and optimization of resources is possible with more collaboration between the communities.

It is also important to establish methods with sufficient statistics, so they be general, which is a challenge due to cost and rarity of measurements on real nuclear material. This highlights the need for international collaboration. IAEA has initiated activities on spent fuel characterization [1].

2. FUEL CHARACTERIZATION

The implementer and operator have to determine to the best possible extent the key characteristic parameters of each individual spent fuel assembly in order to be able to design and operate a safe and economically optimized disposal system of the spent fuel. This is planned to be done through a combination of known fuel history, measurements and codes. These parameters include the following:

— Decay heat;
— Burnup (BU), Initial enrichment (IE), Cooling time (CT);
— Criticality – multiplicity;
— Radiation: primarily gammas and neutrons;
— Nuclide inventory.

For international safeguards verification by IAEA and in Europe Euratom, a number of fuel parameters are verified. These are partly the same as for operational use, a potential issue which is further discussed below.

— Safeguards verification:
  - Identify correct fuel;
  - Missing pins - completeness of fuel assemblies;
  - Contents of fuel – amount of fissile material;
  - Weight.

The parameters are planned to be determined by a gamma and neutron measurement system in conjunction with the encapsulation process together with modelling codes and known history and properties of the fuel assemblies. Due to the large amount of fuel assemblies to be measured, high through-put and robustness of the methods and instruments are paramount, as is the capacity to make fast decisions made based on the measurement results and codes.

2.1 Decay heat

The decay heat is a fundamental property of any spent fuel activity. It is often the limiting factor, which means it has implications for safety as well as economy. It changes as the content of the fuel decays, and in the very long term it will almost disappear. Often the so-called thermal pulse is considered gone after 1000 years.

It is important in all parts of the back end system, such as transportation, drying, intermediate storage (wet and dry), and final disposal. There are typically temperature requirements, typically highest allowed temperature. There are potential issues where a certain temperature interval could be of concern, such as Delayed Hydride Cracking (DHC). There can be situations where the total deposited amount of energy (heat times time) in a certain material or volume is of interest. There can also be requirements on the maximum decay power itself.

There are two basic modes of intermediate storage, dry and wet. In dry, the fuel is stored in casks, and there is a maximum allowed amount of decay power in each cask. Uncertainties are fairly prominent as the few fuel elements in each cask gives statistical uncertainty.

For wet storage the decay power has to be known in order to cool the pools sufficiently, and the total amount of power in the pool will determine the timing of severe events such as loss of cooling of the pool. As there are so many assemblies in a pool, the total uncertainty is small, but the bias is very important.

A final geological repository is passively cooled by non-flowing processes in the rock, which is an inefficient way to perform cooling. As an example, the Swedish final repository will in total have about 10 MW of decay power (about the same as a research reactor) but over a large volume of almost a cubic kilometer. Typically, there are temperature requirements on the canister, the bentonite and sometimes the rock (and perhaps in some circumstances the fuel itself) to be fulfilled. Of particular high economic importance is the potential for optimization of the design of the repository. Examples include amount of fuel in each canister and the distance between deposition holes and deposition tunnels. These cost savings are potentially very high, in the order of billions of Euros. Although different types of repositories are different, they will all have some temperature requirements. A typical example in the Swedish concept KBS-3 is 100°C in the bentonite.

Other important uses of decay power are as a fundamental verification parameter in codes such as Scale/Origin, where nuclide content (e.g. U, Pt), radiation, multiplicity etc. are determined.

It can also be noted that it is an important parameter to evaluate in reprocessing as a large part of the spent fuel's decay power is included in the 'waste' portion of the result of reprocessing, and the footprint of a geological repository may be almost as large as if the fuel had not been reprocessed.

Finally, it can be employed in safeguards, as it is a parameter considered difficult to falsify.

2.1.1. Calorimetry

The basic method to measure decay power is calorimetry. The fuel is measured in a device like a 'thermos'. One of the few full-scale ones, where entire fuel assemblies can be measured, have been operated at the Swedish Intermediate storage facility, CLAB, since several decades. The results have been used, among other things, to verify several codes. A lot of measurements have been openly published.

Calorimetry has the potential to be accurate; in the order of 2% uncertainty. The problem is that it requires long measurement times for each assembly several days for highest accuracy (and around 10 h for normal use at CLAB). In the case of SKB, around 12 assemblies have to be determined per day in the encapsulation plant. This would then require many calorimeters, in different pools, as if they are in the same pool they interfere. This would be very impractical and uneconomic. Thus, indirect determination methods must be developed to a reasonable accuracy [2].

2.1.2. Indirect methods

An indirect method that has been developed over many years, and particularly the last ones, is to use gamma to determine the decay power. Particularly the cesium and europium peaks are suitable. [3–7].

2.1.3. Blind test on decay power

An international effort to blindly test the capacity to calculate decay power on fuel history, led by SKB and in collaboration with NEA/OECD – with more than 25 participating organizations and groups, using most of the internationally available codes, - is presently underway. 5 spent fuel assemblies from CLAB has been chosen in secret and typical data about these given to the participating groups. The groups then independently determine the decay power. The results will be compared to new calorimetric measurements of the fuels.

The overall aim with the blind test exercise is to:

— Learn more about characterization and decay heat determination of nuclear fuel;
— To evaluate:
  - How accurately available simulation codes can predict the decay heat compared to the measured decay heat;
  - How the different codes predict the decay heat compared to each other;
  - How different levels of detail in the operating history data impact the decay heat prediction.

At the time of writing all groups have not been able to deliver their results, and the results are not yet public. Consequently, nothing about the results can be presented in this paper.

2.2. Radiation

Radiation is another result that comes from these determinations more or less automatically. Radiation is seldom a limiting factor, but obviously a very important factor due to safety. It is very important for several reasons to be able to determine radiation dose (all types) with sufficient accuracy, and with known uncertainties. It is used for radiation protection, design of equipment and shields etc. in all parts of the back end system: transport, intermediate storage (wet and dry), encapsulation and final disposal. It is often assumed that there are considerable conservatisms in the predictions of radiation. This is generally true, but there have been a few recent examples in more than one country where this is not necessarily true.

2.3. Criticality

All nuclear material outside a reactor must not go critical. Normally the criticality is determined with codes for a certain set-up. However, there are some situations where there may be large benefits from checking a single assembly in order to treat it more accurately in the system. It is, just as with radiation, something that comes more or less with the rest of the characterization, as multiplicity.

It must be remembered that in the back end the important parameter, k_{eff}, is not so far from 1 (=critical), as the fuel elements are designed to get critical. Thus, criticality always an issue, however usually not a limiting one. It is relevant for all parts of back end system: transport, intermediate storage (wet and dry), encapsulation and final disposal. As the half-life of U-235 is around 700 million years, the criticality issue does not decay as quickly as radiation and decay heat.

2.4. Nuclide inventory

The nuclide inventory is important for the safety analysis of the geological repository. However, the required accuracy of the determination this parameter is low, as a factor of two often is sufficient. The nuclide inventory is an output of the codes and is an integral part of the characterization in terms of the other parameters, such as decay power.

Part of the nuclide inventory is the safeguardable fissile material such as uranium and plutonium.

2.5. Safeguards

From a safeguard point-of-view, geological repositories are an exception in the sense that the nuclear material is not readily inspectable anymore. This means that there will be strict requirement on verification before disposal.

Several of the parameters that have to be determined are also safeguards related in the sense that these parameters are declared by the owner and operator of the nuclear material. These can then be verified. An important issue for the operation of final disposal system is that these parameters have to be determined in the best possible way. One reason is the direct operational optimization of the system (see above). Another is the long term risk, for example that it must not be reassessed in the future so that the safety and dependability of the disposal is put into question. A very substantial problem with two or more determinations of the same parameters is the adjudication between these. What should be used for operational use? The operator must use the best one. If any are considered sub-par, a fundamental problem has occurred.

Therefore, a joint measurement system is proposed by SKB and the Swedish regulator SSM, used by both operator and authorities and inspecting bodies. All relevant data for the fuel, such as its operating history, initial enrichment, burnup etc. will be used for the best possible determination of the parameters together with the measurements of gammas and neutrons. Calorimetric measurement of the heat will be done on part of the fuel inventory as a way to anchor and verify the determinations. As a result of this investigation most likely results to some extent different from the safeguards accountancy data will be reported. The measurement system and electronic would be put under safeguard seals, and the signals split with data authentication techniques.

In addition, it is also foreseen that Cherenkov radiation devises will be used to inspect the assemblies.

Another issue with verification of declared data is that several of the parameters have been calculated by a certain code version at a certain time. For example, a fuel assembly declared in 1980 could have a different value than the same fuel assembly, with exactly the same operational history, declared in 2019 just because a new code version would be used.

A further issue that has to be considered is the mistakes in the records and data bases – 'true errors'. We find these in the Swedish records, and it is highly likely that they exist in all countries. These are completely non-systematic. In the proposed system they would be found due to the multiparameter approach with would raise flags for abnormalities. The multi-approach determination still gives reliable value for these, although possibly with a slightly larger uncertainty.

But the assemblies with faulty records must be dealt with, and they must be possible to dispose; it cannot be reasonable to suggest that they forever would not be disposed due to non-verification- it may never be possible to resolve these mistakes in some (likely few) cases. It must be better to have them safely disposed underground than not.

In summary it is proposed that at the time of disposal the best possible characterization of the fuel assemblies is done, using the history and properties of the fuels, codes and state-of-art measurements. This determination then represents the future record.

It should be continued to be investigated if it is possible to develop one joint measurement system, which also confirms that no rods have been removed, for both operator and IAEA/Euratom. The continuity-of-knowledge for each fuel assembly will be relied on [8].

2.6. Uncertainties

The concept of uncertainty plays an important role in the strategy. For the safe and cost-efficient disposal of the spent fuel the demands of accuracy and uncertainty for the final verification of the different parameter are typically:

— Decay heat: very high accuracy, order of few percent uncertainty;
— Criticality: very high accuracy, order of few percent uncertainty;
— Radiation doses: high accuracy – maybe 10 %;
— Nuclide inventory: for most nuclides fairly low accuracy need; <100 % (for some nuclides higher accuracy needed);
— Safeguard verification: amount of fissile material, burnup, initial enrichment, cooling time, missing pins: intermediate accuracy.

3. FUEL CHARACTERISATION ACTIVITIES AND INTERNATIONAL COLLABORATIONS

For SKB, various activities aiming at making sure that by the time of start of operation of the encapsulation plant, sufficient measurement methods, codes, fuel data and knowledge and understanding of the nuclear fuel for operational and safeguards issues, and in the end for long term safety are known and developed; and also, that sufficient competent human resources are available.

The planning horizon is around 10 years (approximate start of operation of the encapsulation plant). Important activities are Project Fuel characterization, Project Fuel information, and code development.

These projects develop measurement detectors and systems and methods, and related codes, to be placed in the encapsulation plant and potentially used for all disposed fuel elements.

Several large and important international collaborations exist. There has been since several years collaboration between Sweden and USA – Department of Energy and Los Alamos NL, Oak Ridge NL, Lawrence Livermore NL and Pacific Northwest NL for example. A number of other countries, such as Belgium, Japan, South Korea, Germany, Euratom and the European Commission JRC are also involved. The new European Commission project EURAD ('Joint programming') is just about to be initiated; in EURAD one large work package is devoted to spent fuel characterization. IAEA has activities with consultancy and technical meetings underway, with a policy report as its goal.

The measurements of the so called SKB-50 — 25 BWR and 25 PWR fuel assemblies in CLAB (the Swedish intermediate storage facility in Oskarshamn) — with calorimetry, gammas and neutrons, and other techniques, constitute an important basis for the activities. [3–7]

Another fundamentally important issue is the basic nuclear data and cross sections. The management and assessment of these may be of great importance to the effort. In various collaborations this is also covered, where laboratories, for example, as SCK Mol, in Belgium and Oak Ridge NL play prominent roles.

3.1. Fuel data management

SKB has a special long term project on fuel data management. It aims to preserve sufficient fuel data for all fuel elements to be finally disposed and restore records where applicable. Apart from the records and databases of SKB itself, records at the nuclear power plants, fuel suppliers and laboratories are utilized. It also defines what data that have to be preserved and available for the final repository (and partly for other parts of the back end system).

One conclusion so far is that there is missing and erroneous data present in the records. The extent of this has not been completely determined yet. The implications of this are discussed elsewhere in this paper [9].

4. CHARACTERIZATION OF FUEL INTEGRITY

Damaged or failed fuel must be found and treated before final disposal. It is beyond the scope of this paper to describe the methods to do this; there are a number of commercially available methods. Sweden is carrying out a complete programme to find and treat all its damaged fuel in a way acceptable also for final geological disposal.

The spent fuel from the nuclear reactors will in many cases have a very long history before final disposal, from a few years up to perhaps more than a hundred years. It is known that in singular cases, for example fuel assembly handles have been broken during handling. For the handling of the spent fuel it is essential that the vast majority of assemblies can be handled without issues at least up to encapsulation for final disposal. Therefore, fuels with some known property that enhances the probability of failure should be characterized as potentially problematic. Examples include fuel materials, high burnup and chemistry in storage pools. The potential long storage times mentioned above for spent fuel together with these other potential issues have not been fully investigated yet, but investigations and inspections are continually performed in many countries.

In nuclear chemistry and physics there has for a long time been research on the fuel behavior in different time frames. Large project to be mentioned include the SCIP I–III (the OECD/NEA project Studsvik Cladding Integrity Project; phase four is now under initiation) [10]. The issue will be also included in the European Commission project EURAD mentioned above (part of the joint programming effort by the European Commission, now under contract signing) in the work package on spent fuel characterization [11]. Another European Project, DISCO, investigates dissolution rates and behavior of for example doped fuels and MOX fuel [12].

TABLE 1. TENTATIVE TABLE OF VARIOUS IMPORTANT CHARACTERIZATION PARAMETERS TO DETERMINE IN THE BACK END SYSTEM

Characterization parameter
Decay power
Radiation dose
Gamma
Neutrons
Criticality/multiplicity
Nuclide inventory
Burnup
Initial enrichment
Cooling time
Safeguards parameters
Burnup
Initial enrichment
Cooling time
Amount fissile material
Weight
Cherenkov radiation
Damaged
Risk to integrity
Dissolution rate in water

6. CONCLUSIONS

Fuel characterization is a necessary step in all parts of the back end of the nuclear fuel cycle. The international development of this field is strong, and it is planning to be mature at the appropriate times for the various spent fuel programmes in the world.

In the paper it has been shown how the various parameters necessary to characterize are connected, and how a combination of fuel data, codes and measurements can give determination with sufficient accuracy and uncertainty (Table 1). In terms of economy the decay power parameter is considered the most important, and in most need of development, as it is beneficial for safety and economy if the decay power can be determined with a very high accuracy and very low uncertainty. Fuel integrity has been discussed, and the conclusion is that also properties such as if a fuel assembly is damaged, or if its integrity is at risk in the handling process, should be part of the list of characteristics of the fuel assembly. Also, the fuel chemistry properties should be characterized, such as dissolution rates in water.

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3.7. TRACK 7 – CHALLENGES IN AN INTEGRATED APPROACH FOR THE BACK END SYSTEM

Overview prepared by C. Evans (France), D. Hambley (United Kingdom) and K. Agarwal (India), **Track Leaders**

This track discussed various aspects of the integration of the back end system. Several presentations emphasized on the integration of the various stakeholders involved in the implementation of the spent fuel management programs. For instance, analysis involving all stakeholders, nuclear power plant operators, spent fuel management service providers, waste management organisations, safety authorities and technical support organizations (TSOs), as well as R&D entities to develop ageing management guidance in the U.S. or ensuring robustness of the industrial French fuel cycle were developed. Additional emphasis on the need to include all stakeholders including public communities and administrations since the beginning and maintaining this network while implementing activities is essential. This was illustrated for transportation of spent and recycled fuel from France or siting of centralised spent fuel storage facility.

Another aspect, developed by Russia, was the integration of back end fuel cycle facilities (spent fuel storage, reprocessing, “recycled” fuel fabrication, future partitioning and transmutation facilities as well as final waste storage and disposal facilities) on a unique site, the Mining and Chemical Complex (MCC).

Several papers emphasized on modelling and simulation analysis required to assess various spent fuel management options, including risks and opportunities definitions. These analyses will allow to define mitigation and optimise spent fuel management strategies.

Looking at the current spent fuel stockpile and its future growth, leading to uncertainties associated with the need to implement extended storage period of spent fuel, innovative methodologies, integrating risks and valuing flexibility, were thoroughly discussed. Some for instance allow to develop optimal portfolio management of spent fuel inventories considering all options (direct disposal or recycling options) thus minimising financial risks. Other methodologies include the uncertainties by design of the spent fuel management programs.

Although, spent fuel extended storage period is now a reality shared by various stakeholders on a worldwide basis, it needs to be accounted for, pursuing the development of an end point, i.e. geological disposal of spent fuel or HLW from reprocessing, which was recognized as a key enabler or even a must by all participants to ensure the sustainability of nuclear power.

Finally, collaborative work on international/ multinational management schemes based on the development of shared infrastructures for storage, reprocessing/recycling and disposal was also described as an effort to pursue to overcome challenges in spent fuel management system implementation.

Session 7.1: Challenges in an integrated approach for the back end system

Session Chairs: I. Seelev (Russian Federation) and B. Carlsen (United States of America)

Session 7.1 comprised of six presentations, three from United States of America, one from France, one from Russian Federation, and one from Belgium.

- **Paper ID#99 by R. Stoll (USA)** summarizes the progress and status of the Execution Strategy Analysis (ESA) tool developed by the USA Department of Energy. This tool includes both a subject matter expert elicitation process and a dynamic simulation modelling capability for use in the analysis of alternative implementation strategies and plans associated with an integrated nuclear waste management program. In 2017 the ESA model was further enhanced by developing a stand-alone ESA Origin Sites Readiness Model that represents all the activities and milestones necessary to establish at-reactor and near-reactor site transportation infrastructure.
- **Paper ID#107 by A. Presta (France)** emphasized the importance of and described the methodology and processes employed by Orano TN to engage and ensure effective stakeholder participation in siting of centralized fuel cycle infrastructures such as Consolidated Interim Storage (CIS) and transportation activities. The presentation discussed the evolution of communications tools and provided examples based on maritime transportation of MOX fuels and vitrified residues from Europe to Japan and development of a centralized interim facility in the USA.
- **Paper ID#110 by J. Wise (USA)** summarized the Nuclear Regulatory Commission (NRC) NUREGs 1927 and 2214, which provide guidance for ageing management of spent fuel dry storage systems in the United States of America. The presentation also summarized the Nuclear Energy Institute (NEI) 14-03, developed to complement the NRC guidance and to establish an information database as a mechanism for licensees to share operating experience. Lastly, the presentation noted that the NRC is developing internal procedures to evaluate, through inspection, the storage facilities' performance of their ageing management programs. Lessons learned from NRC Temporary Instruction TI 2690/011 will inform the development of a new NRC inspection procedure.
- **Paper ID#57 by J. Choi (USA)** provided a number of suggestions for addressing challenges in the back end of the nuclear fuel cycle. These include development of multi-national repositories which could decouple the power generation from long term spent fuel management, enhance nuclear safety, reduce security and proliferation risks, as well as provide flexibility and retain options for future strategic changes. Other suggestions included consideration of a high-velocity oxy-fuel spray process for applying a corrosion-resistant neutron absorbing material to fuel packaging components as well as an alternative processing and disposal scheme that would avoid the need for a mined geologic repository as presently envisioned.
- **Paper ID#90 by E. Zhurbenko (Russian Federation)** addressed the importance of up-front national planning for the nuclear infrastructure that is essential for the effective implementation of the nuclear energy programme. Key factors influencing the strategy for selecting an open or a closed nuclear fuel cycle were presented along with the associated impacts on the options for spent fuel management. Options for newcomer countries were presented and the advantages of partnerships with countries with mature nuclear technologies and programs.

- **Paper ID#161** ***(Invited)*** **by L. Van Den Durpel (Belgium)** discussed the economic implications of the uncertainties associated with the back end of the nuclear fuel cycle and the uncertain costs and timing of geologic disposal (and the associated impact on storage duration) versus the economics of reprocessing and recycling schemes. The importance of these implications was discussed, and modelling schemes were presented for translating these uncertainties into financial risks that can support evaluating options. Assumptions, in particular to the discount rate, employed in the models will be dependent of each context and needs to be carefully customized.

Session 7.2: Challenges in an integrated approach for the back end system

Session Chairs: C. Evans (France) and C.P. Kaushik (India)

Session 7.2 comprised of six presentations, two from France, one from Russian Federation, two from United States of America, and one from World Nuclear Association.

- **Paper ID#93** ***(Invited)*** **by J. Czerwin (France)** of ORANO presented "Valuing flexibility and integrity risk in used nuclear fuel management." It emphasised on various aspects required independently of the chosen option (direct disposal of spent fuel or recycling options) like long term interim storage of spent fuel, transportation and deep geological disposal of final waste. Considering specific context of each countries/utilities, evaluation of optimal scenario for portfolio management of spent fuel inventory, based on innovative methodologies, valuing risks and flexibility, using all available industrial technologies including recycling, was presented. There was an interactive participation from the audience.
- **Paper ID#174 by S. Missirian from EDF (France)** presented "French Cycle Impact Approach". Thorough assessment of the robustness of the French cycle implementation for the coming decades completed by industrial stakeholders (EDF, Orano, Framatome, ANDRA) and reviewed by the Nuclear Safety Authority (ASN) was detailed. It was indicated that choices made by industrial stakeholders do not create unacceptable consequences regarding the entire French fuel cycle in the mid-term. The presentation was well received by the audience.
- **Paper ID#178 by I. Seelev (Russian Federation)** presented "Closing the fuel cycle" at Mining and Chemical Combine (MCC). It described the existing and planned infrastructures to be implement at MCC site covering all aspects of the closed fuel cycle: wet and dry storage facilities, the pilot demonstration plant to reprocess spent fuel and future commercial plant RT2, MOX fuel fabrication and separation of HLW, and long lived actinides extraction from HLW and their burning in a Molten Salt Reactor (MSR). There was a good participation from the members during question session.
- **Paper ID#144** ***(Invited)*** **by B. Carlsen (USA)**, Idaho National Laboratory, presented "Facing the reality of indefinite storage of spent nuclear fuel". It stressed that storage duration for spent fule is uncertain due to delays in developing geological disposal facilities. Extended period of spent fuel storage will be required although the exact required duration is not known. This long term uncertain duration has to be accounted for in the design of the spent fuel management programme. Nevertheless, indefinite duration does not state that there is no end point, and efforts to develop geological disposal capacities remains mandatory. There was good discussion among the participants during the questions and answers session.

- **Paper ID#152 by H. Zaccai (World Nuclear Association)** presented “Responsible management of used nuclear fuel by the nuclear industry”. It emphasised that political and social acceptance along with strategy are the main drivers of the used fuel management. Nuclear industry has implemented solution to safely manage spent fuel and will continue its efforts to implement innovative solutions allowing to manage spent fuel up to its disposition. This is a prerequisite to ensure the sustainability of nuclear power and its development to mitigate climate change as spent fuel management is considered by both public and politics as the Achille’s heel of nuclear energy technology. The presentation was well received and appreciated by the participants.
- **Paper ID#119 by A. Bednarek (USA)** from Nuclear Threat Initiative was on “Co-operative spent fuel management partnership in the Pacific Rim”. It mentioned that the partnership is composed of a small group of working level experts on nuclear spent fuel management from several stakeholders with an objective of better understanding and improved relationship among key spent fuel and waste management experts. The presentation was well received by the participants.

Paper ID#93
VALUING FLEXIBILITY AND INTEGRATING RISKS IN USED NUCLEAR FUEL MANAGEMENT

J. CZERWIN
Orano
Paris La Défense –France
Email: john.czerwin@orano.group

C. EVANS
Orano
Paris La Défense –France
Email: cecile.evans@orano.group

Abstract

As the first phase of the worldwide nuclear fleet is now approaching 40 years of operation, the Back end of the fuel cycle is becoming a forefront focus for utilities having to deal with pool saturation, reactor shutdowns, and requirements for extended periods of interim storage following significant deferral in the implementation of centralized interim storage or geological disposal facilities. As generated radioactive by-products are increasingly being seen as the Achilles heel of our industry, implementation of responsible used fuel management is a condition to ensure sustainability and expansion of nuclear as a low carbon energy source. Given the dynamic and uncertain market environment, cost of electricity and financial performance are not only important to historical utilities but are also key for the development of new capacities in large mature nuclear countries, expanding countries or newcomers. In this context, Back end management with its long term liabilities and associated risks has a growing impact on utilities' financial performance and risk, development potential and market value.

Used fuel and related waste management requires an overarching long term multi-dimensional system approach which is implemented in stages. A suite of options could be available over the long term, allowing integrating future informed decisions which provide safe, economic solution mitigating risks and uncertainties could be deployed.

Used fuel management system involves multiple decisions over time encompassing conflicts of drivers, uncertain factors and alternatives arising as the market or environmental conditions evolve. Uncertainty and risks are of different natures: technological, environmental, socio-political, economic and financial. Therefore, flexibility in back end options offers mitigation for the uncertainty of risks. Valuing flexibility and integrating risks when assessing decisions will allow utilities and their stakeholders to decide which option to develop and when.

Orano, providing industrial and innovative back end solutions and services for over 40 years, will share its developments allowing implementing various alternatives to manage used fuel matching a NPP-operator's specific financial cost and risk objectives.

1. INTRODUCTION

While safety of nuclear reactors and their economic competitiveness are strong drivers for the sustainability of nuclear power, management of used fuel and waste are increasingly critical as it concentrates a very wide spectrum of stakes, from non-proliferation and security to long term safety, through environmental impact and public acceptance. As the first phase of the worldwide nuclear program, started in the 1980s, is now approaching 40 years of operation, the back end of the fuel cycle is becoming a forefront focus for utilities having to deal with pool saturation, reactor shutdowns, requirements for extended periods of interim storage mainly dry storage at reactor site following significant delays in implementation of centralized interim storage or geological disposal facilities.

Pressure around used fuel management will increasingly grow, considering the current volumes of already unloaded fuel, and the strong expected growth of both used nuclear fuel inventories from operating plants and shutdown reactors in the coming decades.

Factors to take into consideration are of different natures: technological, environmental, safety and security regulations, public preoccupations and economic & financial.

Given the dynamic and uncertain market perspectives, cost of electricity and financial performance are not only important to historical utilities but also key for the development of new capacities whether in large mature nuclear countries, expanding countries or newcomers. Deferral of decisions leading to extended storage of used nuclear fuel at reactor sites could become a major impediment to the future expansion of nuclear energy due to the inability to implement a comprehensive management solution for spent nuclear fuel from existing reactors.

In this context, Back end management with its long term liabilities and associated risks has a growing impact on utilities' financial performance, development potential and market value.

2. USED FUEL MANAGEMENT OPTIONS

Used nuclear fuel management related to the production of nuclear-based electricity is a major challenge requiring a long term strategic planning including technical plans and methods for the financing of all future actions. It is therefore and rightly so, the subject of special attention requiring a rigorous road map framed by nuclear national law or even transnational such as the European Directive of 2011 for the responsible and safe management of spent fuel and radioactive waste.

So far three different strategies have been adopted for used fuel management:

- **"Closed fuel cycle"**, where the spent fuel is reprocessed and separated reusable materials recycled in nuclear reactor: this strategy has been partially implemented, i.e. mon-recycling of Pu and RepU at industrial scale in various countries. One third of the worldwide discharged fuel so far has been reprocessed [1].
- **"Open fuel cycle"**, where the spent fuel is considered as a waste and is stored on an interim basis pending the availability of geological repository.
- "**Wait-and-see or deferral of decision"**, where no decision has yet been made on a final disposition option and where used fuel is placed in interim storage.

Both open and closed cycle solutions can be considered as sustainable options for used fuel management as they cover all the steps: interim storage in dry or wet solutions, transport, recycling (for utilities/counties having chosen the closed cycle), and final disposal of waste in deep geological repositories.

After 40 years of worldwide experience of the nuclear industry in trying to implement deep disposal repositories, none are yet fully constructed nor operational, although a few countries are making significant progress towards opening a repository, most notably Finland, Sweden and France.

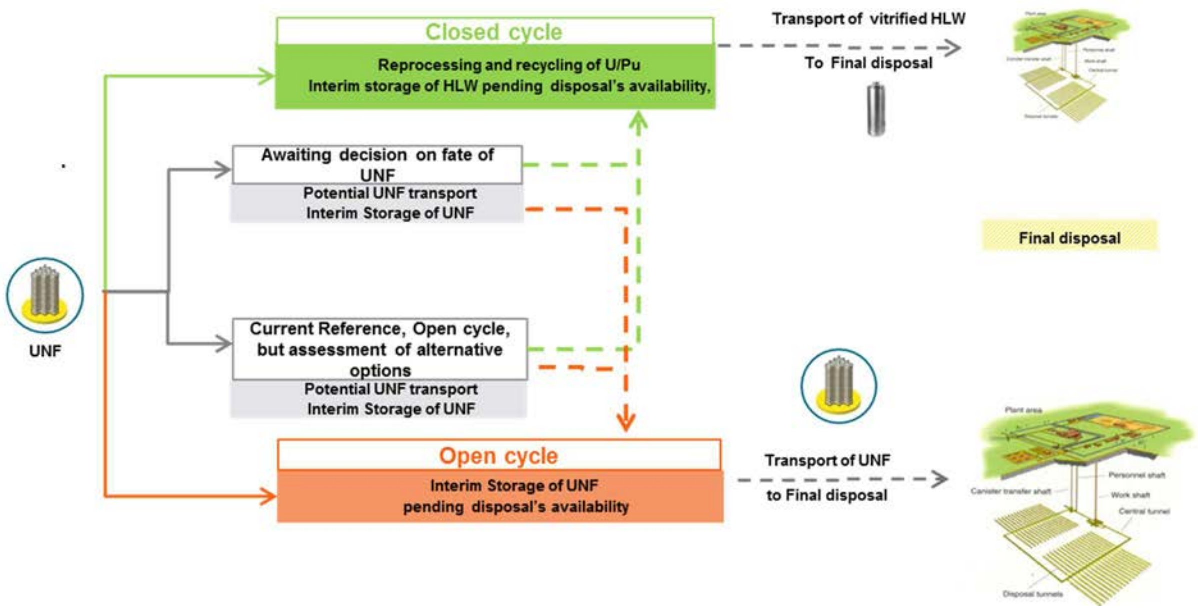


FIG. 1. Strategies for used fuel management.

Therefore, some key challenges arising to implement sustainable used fuel management programmes such as:

- Enhance public acceptance and reduce the cost of geological repository;
- Avoid reactor pool saturation and safely manage interim systems.

Implementing disposal capacities of used nuclear fuel and or HLW involves extremely long planning horizon for siting a repository, construction, and emplacing the UNF or HLW. It remains a complex process, on the technical side (the safety demonstration in particular when fissile material is disposed emplaced in the geological repository) but moreover on the public acceptance and stakeholder involvement.

In addition, continued delays in making decisions for the development of a deep geological repository, transferring the burden to future generations increase both costs and the risk of failure.

Deep geological repository is dependent on the availability of suitable geological conditions in the country and will therefore:

- Remain a scarce resource: optimizing its use is therefore crucial for the durability of the nuclear energy
- Take more time to implement than expected: mastering the elapsed time to their full implementation is crucial to mitigate risks and uncertainties on any intermediate upstream steps in the management of used fuel.

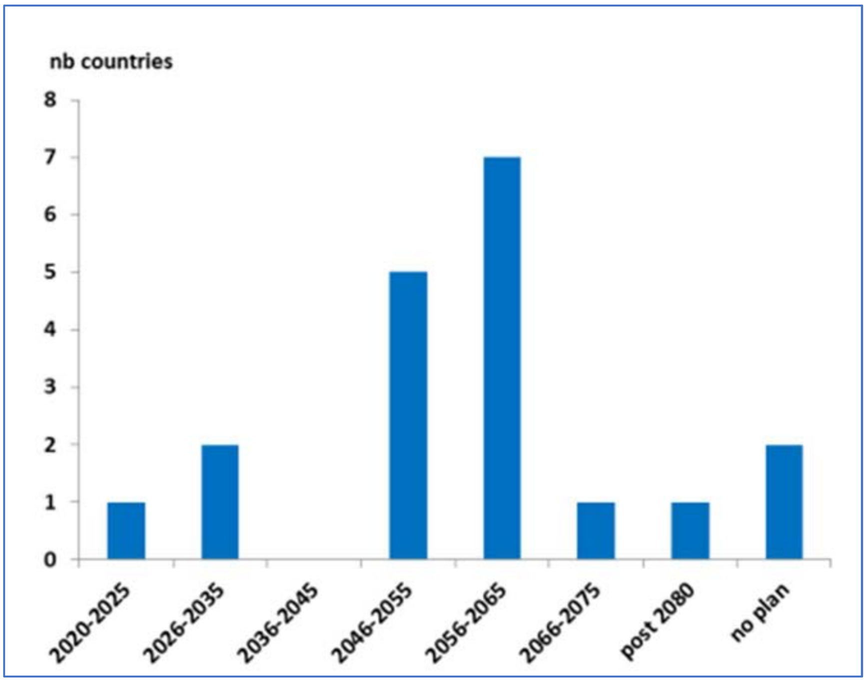
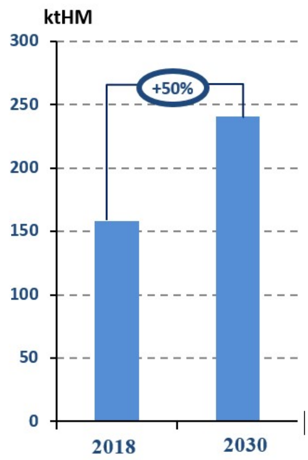


FIG. 2. Illustration of foreseen/forecasted DGR start of operation.

This situation has led to an interim storage over 250 000 tHM of used fuel worldwide; some of it has already been stored for over 60 years. Considering long time operation of the current fleet, there is a need to implement additional storage capacity, wet and dry, beyond those of the existing reactor pools in order to continue safe operation of NPPs. The used fuel inventory will continue to grow significantly in the next decades, taking into account both nuclear new builds and the increasing numbers of shutdown reactors.



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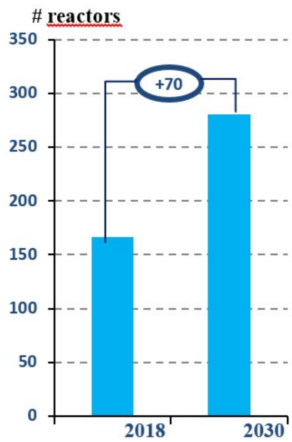


(b)

FIG. 3. (a) Evaluation of LWR used fuel inventories. (b) Evaluation of shutdown reactors.

For all fuel cycles, as the requirements for storage capacity increase, new storage is being built outside of the reactor buildings. Different technologies, as illustrated in Fig. 4, have been implemented or are planned, most of the capacities are using dry storage technologies on the reactor site, but some wet storage facilities are also being deployed.

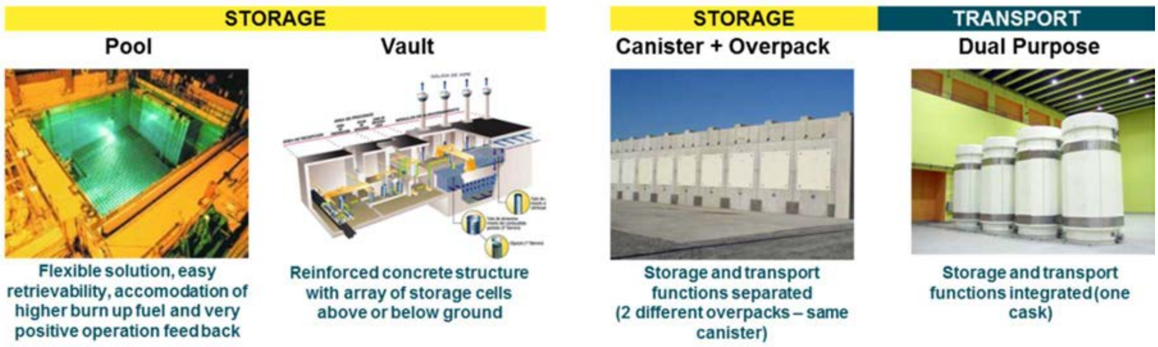
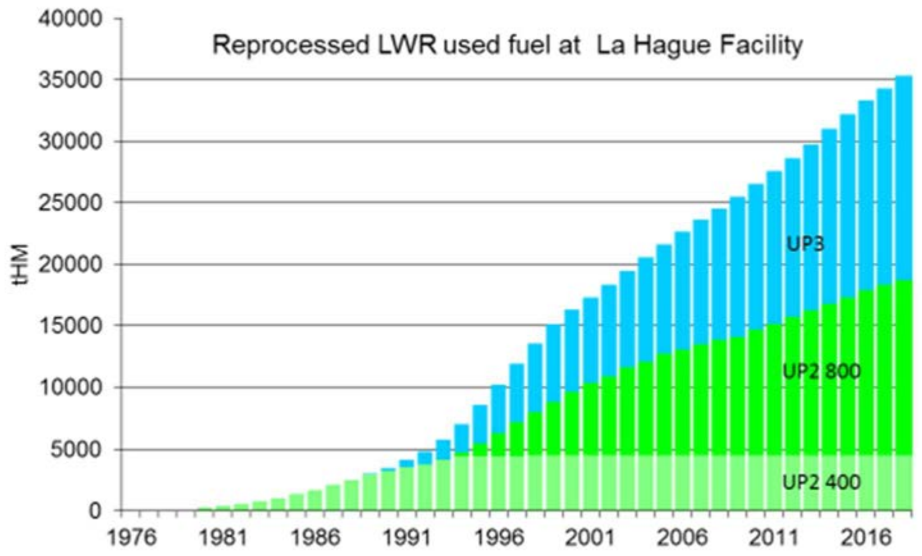


FIG. 4. UNF storage technologies.

The recycling strategies offer many benefits at different steps of the back end of the fuel cycle starting from reactors' pools. Indeed, for countries having chosen this strategy, used fuel are discharged from reactors' pools once cooled down and then transferred to dedicated facilities for recycling. With such a scheme, saturation of reactors' pool is therefore avoided, UNF storage capacity are significantly decreased. Over 100 000 tHM of used fuel has been reprocessed so far. The possibility to have reprocessing carried out abroad is an available strategy choice for any country with a nuclear power programme, since there is a commercial market providing reprocessing services.

For instance, since the mid-70s Orano La Hague reprocessing plants in France has reprocessed over 35 000 tHM of LWR used fuel from nine countries of origins as in Fig. 5. Reusable materials separated at the reprocessing stage have been recycled through MOX and ERU fuel loaded in LWRs. More than 2700 tHM of MOX fuel have been manufactured at Orano Melox facility at the end of 2018. Ultimate High Level Waste is confined in glass matrix, using a high performance vitrification process which is key in the dispositive. So far, 25 000 universal vitrified canisters have been manufactured by La Hague.



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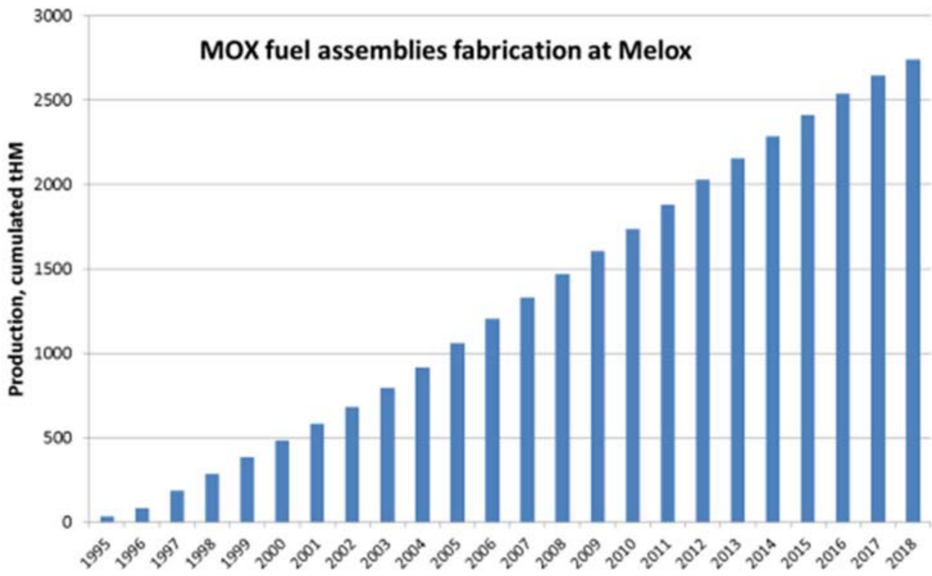


(b)

FIG. 5. (a) Orano La Hague production. (b) Melox productions.

3. A VERY LONG TERM STRATEGIC MATTER WITH RISKS AND UNCERTAINTIES TO MITIGATE

3.1. Storage of UNF and subsequent transportation

All used fuel management schemes require storage of UNF, although the need is considerably alleviated when implementing a UNF reprocessing scheme. For example, in France, storage capacities for over 45 000 fuel assemblies have been avoided.

For countries having chosen the open cycle or deferral of final decision strategy, long term storage is today a reality. Historically, starting in the USA, used fuel assemblies have been stored on-site in dry storage systems designed for 'bridging the gap' for the period of time needed before transportation to the DGR, intended to be for a period between 20 to 40 years. In many cases, there is a need to develop solutions to manage storage facilities, most of the time scattered across many nuclear sites or in the near future stranded storage sites, for extended periods of time exceeding the original design or licensed reference.

The challenge is thus to ensure the long term safety and integrity of both storage system and the spent fuel for many decades to come:

— The SNF and its storage package will require aging management plans to monitor their conditions and include potential mitigation plans should degradation mechanisms occur. This need is already under development in the USA with requirements defined by the regulators and programme under implementation by industrial storage systems providers.
— The challenge goes beyond the storage facility itself. Eventually, the SNF will have to be transported from its initial storage location to a consolidated storage or a disposal facility. Transportation options and safety requirements may change over extended periods of time leading to having to recondition/repack UNF to ensure that the fuel in its future state is exported in packages suitable for transport.
— This may become even more challenging for 'stranded sites' with no capabilities for handling fuel as reactor ponds have ceased operation. Indeed, options for detecting and mitigating potential problems during storage will change as fuel-handling capabilities are decommissioned at former reactor sites. Opening storage canisters at decommissioned sites for reasons such as repackaging fuel to replace damaged or degraded canisters or in response to future requirements on transportation may require building fuel-handling capabilities at the site.

Physical security requirements at storage sites will also change through time. Over longer periods, the dose rate at the surface of the canisters associated with the decaying radioactivity of the fuel decreases exponentially. With continued storage for many decades, stored UNF becomes more vulnerable, its self-protection through its own radiation decreases with time, its self-protection may not provide sufficient protection level against diversion, theft, or sabotage resulting in a need for increased security measures. Those measures will be required for decades, and for stranded sites, even after the nuclear power plant has been decommissioned.

3.2. UNF Encapsulation and deep geological disposal

Encapsulation refers to the placement of the used fuel into robust engineered barriers designed to protect against leakage during long term disposal. Such facilities are even for the most advanced countries such as Sweden and Finland, still under design phase, with some challenges to overcome in terms of fuel characterization to serve both safeguarding requirements and optimization of heat load of final disposal canisters impacting the footprint of the GDF.

Some issues remain with safety demonstration of final disposal canister durability. A widely accepted principle for addressing the long term safety of radioactive waste and used fuel in a geological repository is that the isolation systems should be passive. However, used fuel emplaced in geological repositories is subject to safeguards, and obligation to implement safeguards remains after the repository's closure.

Reprocessing/recycling schemes provides engineered final waste with high confining performance, an additional barrier with demonstrated longevity of the waste integrity. Vitrified wastes have demonstrated capability for very long term safe interim storage and subsequent transportability. They can be stored in demonstrated, well-designed buildings, easily implementable for limited costs and that guarantee the needed radiation protection with passive surveillance only.

In addition, taking out safeguarded materials from the waste allows significant reduction of final waste toxicity and volume. This leads to lower the footprint of the final repository, decrease the complexity of design and final waste emplacement operations and eliminates safeguarding requirements including those after repository closure. These features can benefit the global public acceptance necessary for the deep disposal implementation.

Management of used recycled fuel will differ with various recycling schemes, i.e. starting from a storage phase pending recycling or direct disposal. Monorecycling strategy is most of the time the first step towards implementation of a multirecycling strategy. Nevertheless, in case advanced cycles are not pursued, used MOX and ERU used fuel will have to be disposed of in the GDF, leading to an anticipation of adequate corresponding needs.

3.3. Used Fuel management programmes

The complete programme to implement responsible management of used nuclear fuel and safe disposal of nuclear waste will require a period of over a century. Looking at the main options, some steps are already industrialized, some are still to be deployed and this could extend considerably the overall duration of the programme. Thus, this entails an evolution of risks and uncertainties over time which may be different for the two main management options. As illustrated in Fig 6, uncertainties of open cycle will increase significantly as interim storage of used fuel is extended pending final disposal availability. Recycling options, while dealing with used fuel will also have uncertainties associated with long term storage of final waste storage pending GDF availability, however those uncertainties are lower due to the intrinsic characteristic of this engineered waste form.

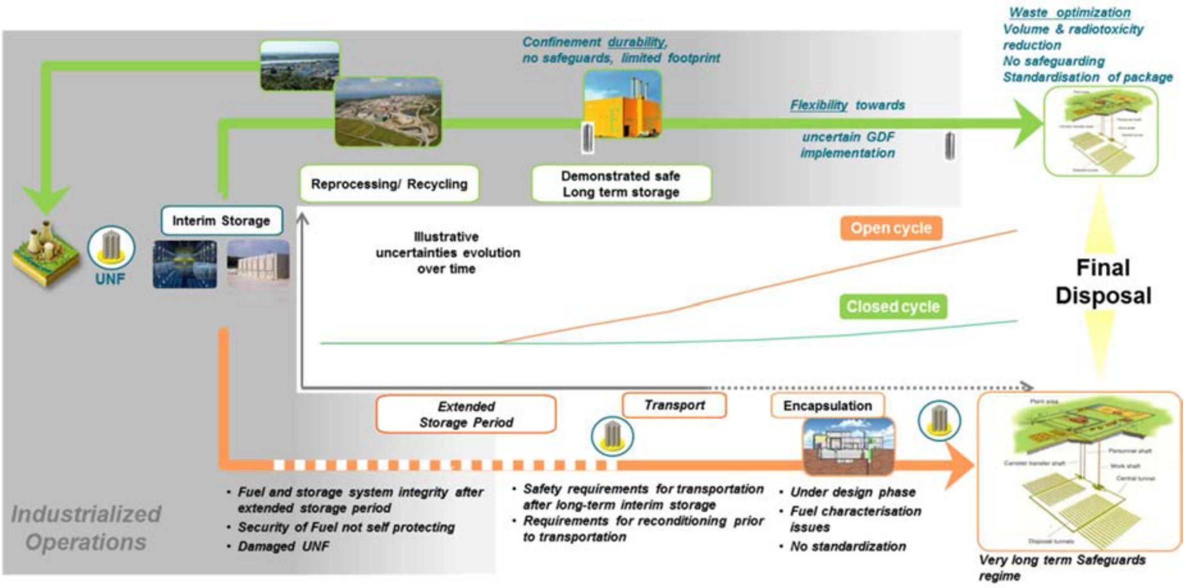


FIG. 6. Used nuclear fuel management: uncertainties evolution.

4. FOR UTILITIES AND GOVERNMENTS: HOW TO DECIDE IN AN UNCERTAIN FUTURE

Regardless of the selected cycle option, financial requirements have to cover all operations for the complete management of spent nuclear fuel up to the final disposal of radioactive waste, and even beyond the closure of geological disposal facility. It encompasses short-term implemented stages but also in a holistic approach the long term ones, given the uncertainties including financial rate of return volatility (including current negative rate of return in some countries). Used fuel management programme should include clear plans for storage (on-site and centralized), transportation to reprocessing plant or final disposal facility, and emplacement of used fuel or HLW in a GDF. These activities are highly interdependent and require putting in place an integrated system approach.

Deferral of programmes and evolution of costs have different impacts on financial requirements depending on the actual nuclear programme status and its remaining time to operate, and the spent fuel management implementation status.

The initial stage, i.e. implementing interim storage, most of the time at reactor site, is often seen to require low financing resources. All of additional requirements for the downstream steps after long term interim storage mentioned previously will add to the costs of SNF management and will increase over extended periods of time.

In the long term, development of GDF and implementing of specific safeguards monitoring regimes will require significant financing with no defined end period.

Throughout the required programme, there are a limited set of strategic decisional points allowing choosing between alternatives. Influence can be made on when to trigger a decision milestone, for example, implementing dry storage systems at reactor site will lock decision on the corresponding inventory for a few decades. The decisional times, td_i as illustrated in Fig. 7; are not continuums but should be decided upon such that maximal flexibility is provided in light of uncertain future developments in all of the UNF management options.

The definition of UNF policy and corresponding programme implementation will require addressing questions such as:

— At the UNF batch level, with analysis of cost/risk balance trade off with some specificities on certain fuel conditions (for example management of damaged fuel).
— Considering overall inventory of UNF exiting and still to come, analysis of the most affordable portfolio management and potential future issues looking at the most attractive alternatives, the time to trigger as well as conditions to execute those.
— Thus, finally being able to establish the programme implementation strategy and securing a critical path for implementing an optimized used fuel management solution and guide development efforts to reduce uncertainties/risks (when is the best moment to contract an option in terms of capacity and duration).

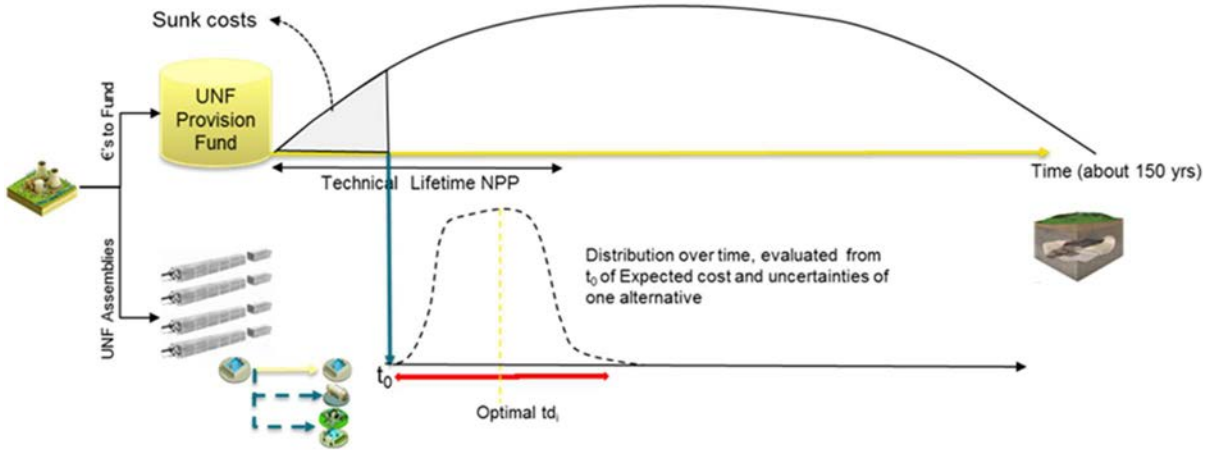


FIG. 7. Illustration of decisional time over the overall used fuel management programme.

For historical utilities with short remaining lifetime of reactor operation, uncertainties associated to used fuel management leads to potential significant increase of required provisions or fees to cover used nuclear fuel inventories management. These situations may translate in significant financial risks and therefore trigger decisions from utilities or stakeholders to mitigate exposure when revenues from operation are limited or have even ceased.

5. DIFFERENT RISK /COST EXPOSURE OVER TIME WITH DIFFERENT FUEL CYCLE OPTIONS

Various fuel cycle options have different cost profiles but also different risks and uncertainties evolving over time: open cycle showing low short-term requirements but uncertainties increasing with time, close cycle options having higher short or mid-term requirements allowing securing long term operation, thus reducing future uncertainties. Considering open cycle and more specifically dry storage at reactor site, although the significant financial requirements may seem to be far away, risks and uncertainties associated to the very long term planning increase significantly with time as shown in Fig. 8. Indeed, critical stages of transport, conditioning in final disposal canister and disposal operations could require additional unplanned financial means corresponding to an evolution of security and safety requirements, ageing management programmes of storage systems, repacking requirements prior to transport, consolidated storage awaiting final geological repository siting, future significant use of potential scarce resources such as copper.

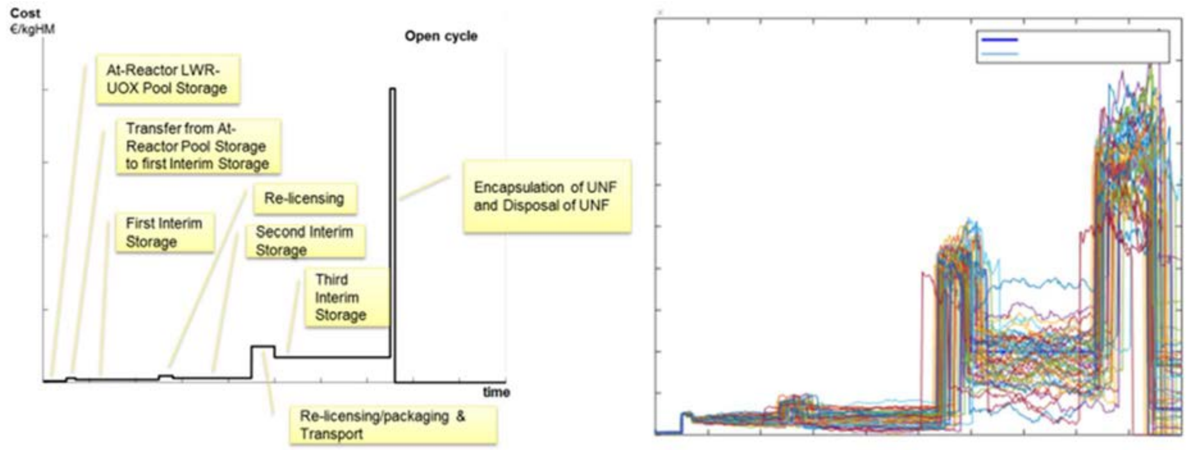


FIG. 8. Nominal Expected Cost-Curve over time and uncertainties applied cost curve for open cycle.

When evaluating /revaluating the programme implementation strategy at a decisional point, the value of various available alternatives including pursuing open cycle or wait and see, but also switching from interim storage towards recycling will be assessed over time considering the combination of costs and risks.

Illustrated in Fig. 9 is an example considering the comparison between open cycle (dry interim storage) and mono-recycling.

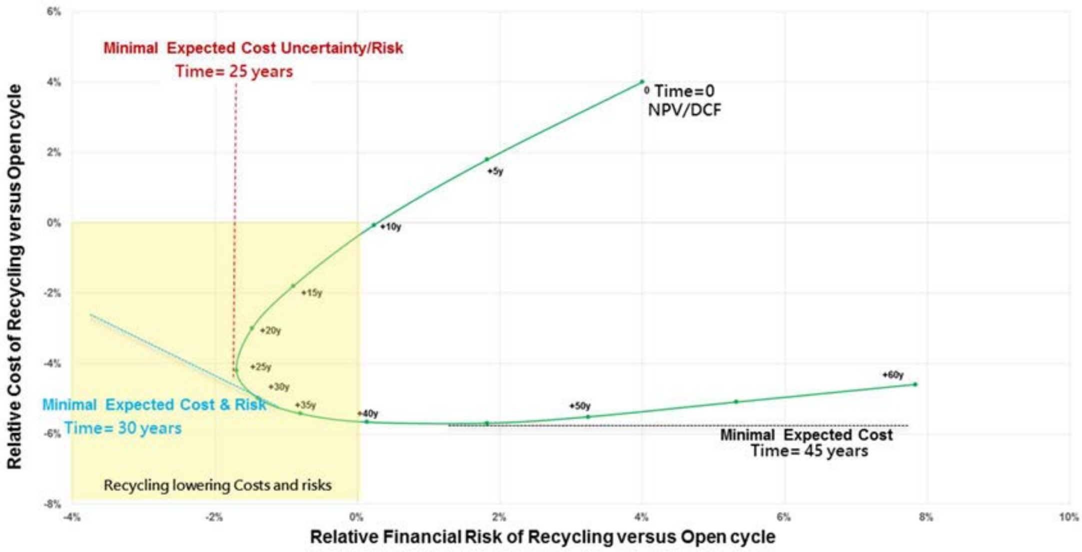


FIG. 9. Illustration of relative cost/risks of alternatives including mono-recycling versus open cycle over time.

The analysis integrates a wide range of assumptions and or assessments such as expected cost of each step of the compared two options, associated uncertainties both in time and cost and financial assessments (discount rates, financial rate of return). It needs to be customized to each specific situation.

Such analysis allows the evaluation, as seen from today (time t_0), of expected cost and uncertainties for the existing and forecasted UNF inventory comparing different fuel cycle options and switching moments in time.

At t_0 the comparison naturally corresponds to the NPV/DCF method. When projecting switching at later decisional times, td_i, the cost of the two alternatives are similar then differing from this point onwards:

the recycling option will limit exposition to known and unknown consequences of long term dry interim storage at the reactor site.

Indeed, reprocessing leads to less amounts of used UOX to be exposed to increasing uncertain requirements from extended interim storage and downstream steps up to GDF costs while providing an option with higher cost predictability. Thus, switches from 10 y from today, up to 40 y, lowers the total cost as viewed from today (including discounting) compared to carrying on with the open cycle implementation.

However, waiting longer may further expose to interim costs and GDF costs uncertainties of used fuels. Thus, the exposition to the consequences of interim storage will be more important compared to those at an earlier decision switching point (this is due to Pu quality decrease impacting reuse performance and Used MOX fuel disposal assuming an define opening window for the GDF).

4. HAVING FLEXIBILITY IN THE BACK END OPTIONS OFFERS MITIGATION TO KNOWN/UNKNOWN RISKS

Used fuel and related waste management requires an overall long term multi-dimensional system approach however implemented in stages. A suite of options could be available in time. Pending future informed decisions, a safe, economic, mitigating risks and uncertainties solution can be deployed. Used fuel management system involves multiple decisions over time encompassing conflicts of drivers, uncertain factors and alternatives arising as the market conditions evolve.

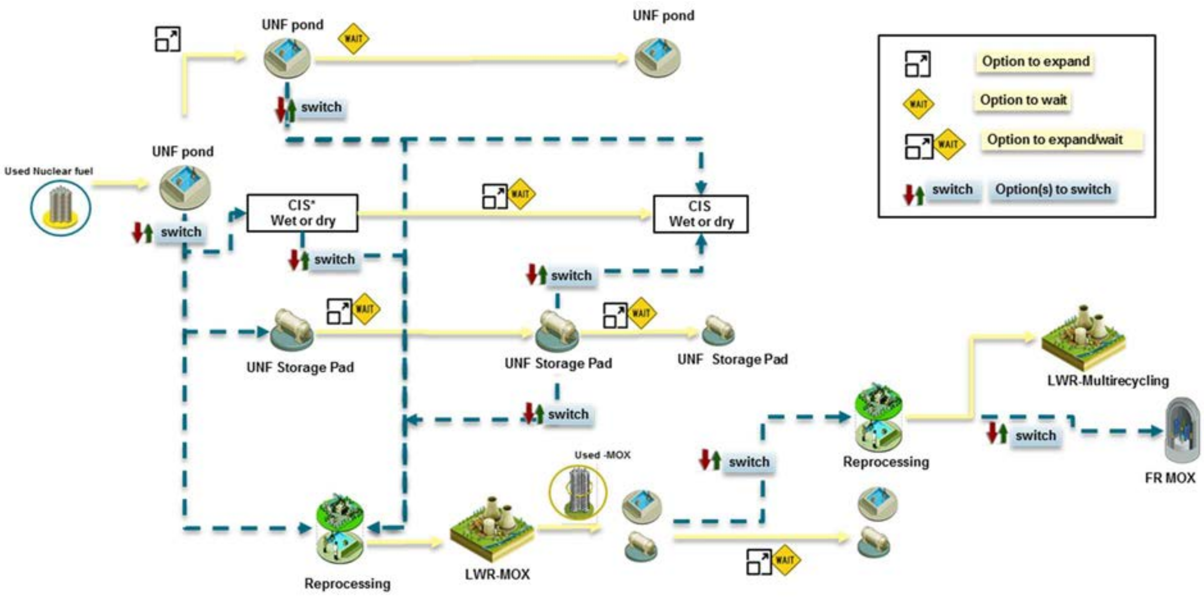
Uncertainty and risks are of different natures. Some are technical- economic such as evolution of commercial service cost or investment overruns, maturity of technology encapsulation and disposal facility and consequences of extended storage of UNF. The evolution of natural uranium market price is also a factor, directly affecting the competitiveness of Gen IV technology and as such the horizon of its commercial deployment which is not currently foreseen until the second part of this century. Beyond environmental, socio- political (evolution of installed nuclear capacities, security and safety requirements, social acceptance of UNF management options,), and financial (discount rates, financial rate of return and utilities competitiveness) aspects will also play significant roles in the optimum route to implement.

International R&D cooperation is undergoing on advanced cycle and reactor technologies and for GDF developments. The nuclear industry is also continuously developing services and solutions to provide additional alternatives to nuclear reactor operators and their stakeholders. Typically, Orano's focus is to improve our reprocessing /recycling services to enhance our scope of services and enlarge our capacities/capabilities based on the lessons learned from our extensive industrial experience and driven by both R&D and innovation efforts. This leads to extend our capabilities to reprocess various types of UNF including VVER fuels from HBU fuels, damaged fuels. Concerning recycled fuel and more specifically MOX fuels, efforts are pursued both on the fabrication process improvement and on fuel developments to increase current MOX fuel performances and develop MIX and CORAIL concepts allowing to multi-recycle Pu in LWR, thus bridging the gap with the future development of Gen IV reactors. This allows offering a wide range of alternatives using existing shared industrial reprocessing/recycling capacities to various countries. Additionally, collaborations with various countries to develop their indigenous reprocessing-recycling capacities are ongoing. To manage at best the potential risks of long term interim storage, new storage systems such as NUHOMS EOS or MATRIX encompassing ageing management requirements by design, reduced footprints or consolidate interim storage infrastructures under development will allow shifting the interim storage step paradigm from a "commodity" to "critical interim system".

The back end of the fuel cycle presents various options branching into a multitude of scenarios Therefore, flexibility in back end options offers mitigation for known/unknown uncertainties and risks.

All these industrial developments provide flexibility in the used fuel management programme allowing minimizing risks and containing costs thus enhancing overall financial predictability of the comprehensive used fuel management solution.

There is no fixed scenario as alternatives arise with new development and the evolution of market conditions. Each decisional branching point involves a value to wait, to extend (in capacity or in duration, for example relicensing of dry storage systems), or to switch options as in Fig. 10.



**CIS: Consolidated Interim Storage*

FIG. 10. Illustration of alternatives to manage used fuel.

The future value of an option is also depending on the developments made to keep the option open, i.e. the value to lose or gain if you do not have the option anymore available.

Using an innovative assessment methodology such as Real Alternative Valuation integrating costs, risks, time and options to determine optimal fuel cycle option can guide utilities and their stakeholders to decide which option to develop and when, thus allowing to secure implementation of a future risk-mitigating strategy.

Multi-capability industrial service approach developed by nuclear fuel cycle operators such as Orano will allow to implement a customized used fuel management programme matching each NPP-operator and stakeholders' specific context and financial cost/risk objectives in dynamic and uncertain market environments.

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FACING THE REALITY OF INDEFINITE STORAGE: WHAT DOES IT MEAN?

B.W. CARLSEN
Idaho National Laboratory
Idaho, USA
Email: brett.carlsen@inl.gov

W.M. HARTMAN
US Department of Energy
Washington DC, USA

J.J. JARRELL
Idaho National Laboratory
Idaho, USA

Abstract

The paper encourages not only acknowledging but pro-actively addressing the issues and opportunities resulting from the uncertainty relative to how and when sufficient repository capacity becomes available. It draws on the work of a former IAEA consultancy tasked with addressing very long term storage of spent nuclear fuel (SNF) as well as current strategic planning and associated initiatives within the U.S. Department of Energy to address the uncertainty in spent fuel management. The paper advocates addressing this uncertainty by design rather than by default. Approaches are suggested for rethinking the basis of spent nuclear fuel storage equipment, facilities, regulatory framework, and communication strategies to acknowledge and proactively address uncertainty relative to the storage duration and the end state of spent nuclear fuel.

1. LOOKING BACK

A look back over a relatively brief 75 years to the beginnings of the nuclear industry offers several valuable lessons that should guide present and future choices. Notably, hopes expressed in phrases like "*too cheap to meter*" [1] and "*the price of nuclear fuels being so low that only hydroelectric power, which is produced without any cost for fuel could compete with it*" [2] have not been realized. In context of an era when "Miss Atomic Bomb" and other nuclear-themed showgirls were Las Vegas attractions [3], atomic science kits were marketed to children, and the Ford Motor Company invested heavily in not just one but two concept cars to be powered by an on-board reactor [4], these predictions did not seem too far-fetched. **Nonetheless, these and other hopeful predictions of a bright nuclear future that seemed reasonable at the time are now generally viewed as having damaged the credibility of the industry.**

The early optimism for the future was not without apprehension. The rapid advance in the technology was focused on the race to produce nuclear weapons. The world's introduction to the unimaginable energy density of a nuclear fuel was in the context of war, with the explosions of the first atomic bombs over Japan. Recognizing the potential dangers of the proliferation of this technology for military and other non-peaceful uses, the "Atoms for Peace" programme proposed another vision: the peaceful controlled distribution of nuclear technology to all countries of the world. In exchange for this potentially life-changing knowledge, countries would agree not to pursue atomic weapons [5]. From this vision, the IAEA was established to encourage international cooperation in channeling this power towards peaceful uses. In the enthusiasm that ensued, nuclear technology was shared freely with many countries, the commercial power reactor industry began to flourish, and the United States Atomic Energy Commission predicted that, by the turn of the Twenty-First Century, one thousand reactors would be producing electricity for homes and businesses in the USA alone [6]. Unfortunately, the nuclear arms race continued, and reactor accidents such as those at Three Mile Island and Chernobyl had additional negative impacts on a public with growing concerns. Reactor construction costs rose sharply, and orders dropped. In a period just over 30 years, the early dramatic rise of nuclear power went into an equally meteoric reverse [7]. **The envisioned nuclear future did not come to pass.**

Nonetheless, over 100 reactors were constructed in the U.S., all of which began construction prior to 1977. These reactors have provided nearly 20% of U.S. electrical production for the past several decades [8]. However, implementation of policy for the management of the spent nuclear fuel (SNF) resulting from reactor operations has proven to be problematic. Early efforts to establish a commercial reprocessing capability in the U.S. have been abandoned as a result of both economic and political concerns. A plant at West Valley, New York, was operated successfully from 1966 to 1972. However, escalating regulations required plant modifications that were deemed uneconomic. A plant built at Morris, Illinois was declared inoperable in 1974 after new technology that, although proven on a pilot scale, failed to work successfully in the production plant. A third plant at Barnwell, South Carolina, was aborted due to a 1977 change in government policy that ruled out all U.S. civilian reprocessing as one facet of U.S. non-proliferation policy [9].

Further, the Nuclear Waste Policy Act of 1982 directed the United States Department of Energy (DOE) to identify a deep geologic repository for disposal of SNF and high level waste (HLW) and to take custody of SNF from commercial reactors beginning in January 1998. This did not occur, and in 2010, with over $15.4 billion invested and a license application under review by the Nuclear Regulatory Commission, the DOE decided to terminate the Yucca Mountain repository programme because, according to DOE officials, it was not a workable option and there were better solutions that could achieve a broader national consensus. DOE did not cite technical or safety issues [10]. Notably, the NRC was directed to complete its review in 2013 and in 2015 reported that the application satisfied nearly all of its regulations. Also, in 2015, DOE announced plans to build two repositories, one for most of the nation’s defense-related radioactive waste and another for commercial SNF and residual defense waste. In 2016, the National Defense Authorization Act denied funds for a defense-only repository. The social and political opposition to a permanent repository, not the technical issues, have proven to be the key obstacle to moving forward with a repository [11]. **Once again, the future did not unfold as predicted – and future prospects for permanent disposal remain unclear.** Amidst this uncertainty, several states have implemented restrictions on the construction of new nuclear power plants [12].

2. LESSONS LEARNED

Looking back, the key lesson seems to be that the nuclear industry has not been well served by building policies and infrastructure around what seemed, at the time, like a reasonable prediction of the future. The path taken has resulted in costly missteps and changes in direction. But, perhaps more importantly, the unrealized expectations have damaged the trust of key stakeholders who were led to believe that a final solution would be in place long before now.

National policy must be decided and implemented before SNF (and HLW) in storage can begin moving to reprocessing or disposal. Given the political challenges that have been experienced in siting a repository coupled with the relatively low near-term costs and risks associated with continued storage, it is likely that SNF will remain in storage much longer than originally envisioned, perhaps many decades or longer. Acknowledging this fact suggests a new framework for planning, design, operation, and regulation of SNF storage (SFS) and associated systems [13].

Past planning and infrastructure for management of spent fuel was based largely on a presumed future that has not occurred – at least not within the timeframes expected. Because a decision taken today could foreclose a transition to another step tomorrow, one of the main challenges is to maintain enough flexibility to accommodate the range of potential future options for the management of spent fuel given the current uncertainties regarding storage duration, future technologies, and future financial, regulatory, and political conditions.

Accepting and accommodating uncertainty invokes a need to explore potential future scenarios with the objective of developing robust facility and equipment designs as well as regulatory strategies that perform acceptably not just for the predicted future but under a broad range of potential futures. Indefinite storage is a future scenario that should be considered. The term indefinite is not intended to mean “forever.” It simply means that storage durations cannot be defined with certainty. Although indefinite storage has generally not been considered acceptable, the fact is that, due to the dependency on unpredictable future events, storage durations are unknown. Openly discussing this reality has proven controversial, largely because it is not consistent with existing expectations.

In hindsight, the presumption that one must know the future to effectively manage spent fuel was neither reasonable nor necessary (nor even possible). Until available disposal and reprocessing capacity exceed the rate of spent fuel generation, spent fuel inventories and average storage times will continue to grow. It is important to recognize that neither increasing SNF inventories nor longer storage times necessarily pose an unacceptable situation. Spent fuel storage systems and equipment can be monitored, maintained, and retired and replaced when deemed appropriate. This is not unlike the way the airline industry; buildings, bridges, and roads; and other major capital equipment are presently managed. Further, once this is recognized and accepted, spent fuel storage facilities and systems can be sited and designed to facilitate this.

Managed Options for re-use of spent fuel are preserved at relatively low cost while in storage. Preservation of options for future fuel cycle choices has been undervalued in the debate about fuel cycle policy. SNF can be safely stored at reactor sites, centralized storage facilities, or geological repositories designed for retrievability (an alternative form of centralized storage).

Storage of SNF, for as long as it takes until society selects and implements an end state, need not be considered unmanageable nor unsustainable. Consider that the ~80 000 tonnes of SNF generated by the U.S. more than any other country, would fill a U.S. football field (i.e., 49 m × 110 m) about 20 m deep[3] [11]. Contrast this with a block of coal 1560 km high on that same field to produce the energy equivalent[4]. It should be noted that, when burned, coal is converted to solid waste in the form of ash or gaseous waste in the form of emissions.

The intense energy density of nuclear fuel can be considered an asset not only because it drastically reduces CO_2 and other greenhouse gas emissions relative to energy from fossil fuels but also because it enables its by-products to be contained in a manageable volume rather than dispersed into the environment. SNF is considered as an asset rather than a waste in some countries today and, as technologies evolve, future generations may use it to fuel advanced nuclear fuel cycles or may find other beneficial applications for this nuclear material. This, along with an enviable record of safe storage and transport, is contributing to an emerging paradigm for spent fuel management that is based on preserving flexibility while ensuring safety and security and without foreclosing options or passing an undue burden to future generations.

3. LOOKING FORWARD

The nuclear community is now faced with an opportunity stemming from the recognition that SNF disposal solutions are likely decades or more away – coupled with a concurrent recognition that nuclear power can play a substantive role in addressing pressing environmental and societal needs. Figure 1 below, based on information from the 2018 edition of the IAEA's energy estimates through 2050 [15], projects that nuclear power generation could range from a decline of ~10% to an increase of ~90% over the next 30 years.

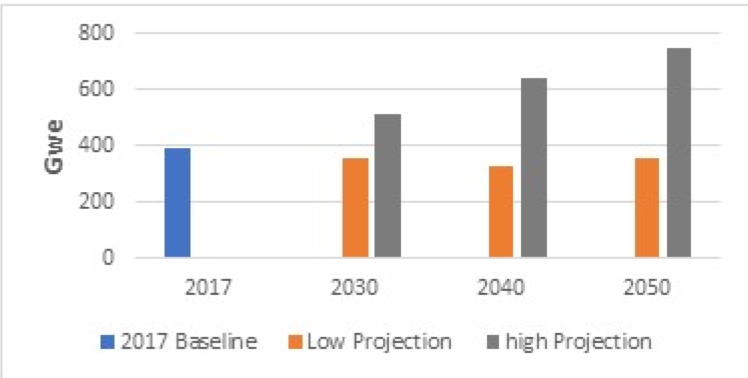

FIG. 1. World Nuclear Electricity Projections.

[3] Comparing this estimate of 20 m deep with a former GAO estimate of a football field 15 feet deep based on 65 000 tonnes leads one to believe that 80 000 tonnes would fill the football field only 20 feet (~6 m) rather than 20 m deep. This has been independently confirmed by the author.

[4] Based on 45 GWD energy per metric tonne SNF and 10^6 BTU energy per cubic foot of coal. Note that when burned, coal is converted to solid waste in the form of ash and gaseous waste in the form of emissions.

With a significant amount of SNF currently in storage, no SNF disposal facilities presently available, and world-wide reprocessing capacity well below current the rate at which SNF is being generated, even the low projection can be expected to result in increasing SNF inventories along with the attending increase in storage durations. Figure 2 based on information from Table A-14 of [16] summarizes the current projections for the status of SNF inventories from 13 countries.

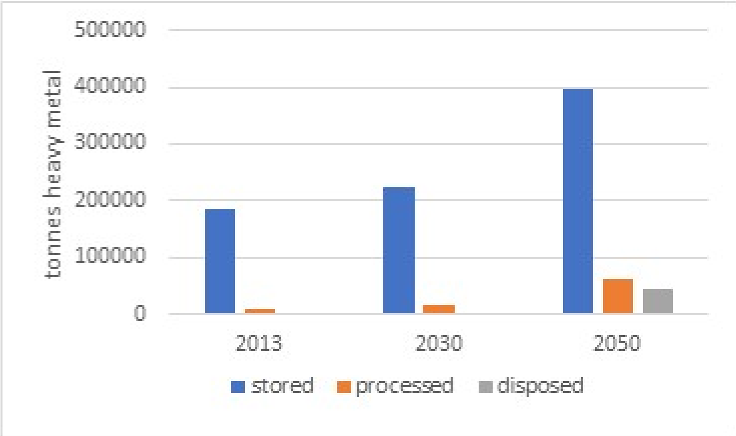

FIG. 2. Spent fuel inventory projections.

The high projections for the nuclear energy contribution would be expected to further increase inventories and storage durations. Interestingly, the high projection presumes that nuclear energy maintains only a constant ~5.7% share of a total energy market that grows by ~90%. This is significant because growth in the nuclear market share should also not be ruled out. History confirms that public attitudes toward nuclear can swing substantially in relatively short periods. Additionally, past projections of energy needs and the energy mix have not been particularly good at matching future reality. Much could change in the energy landscape as the third world develops, the transportation sector is electrified, and environmental and climate impacts are better understood.

Nuclear energy could play a significant role in addressing one of the key challenges facing the twenty-first century – dramatically reducing emissions of greenhouse gases while simultaneously expanding energy access and economic opportunity to billions of people. Without a substantial contribution from nuclear energy, the costs of achieving the necessary deep decarbonization of electricity increase significantly [17]. In other words, it is plausible that the nuclear future may not unfold as currently predicted and that nuclear energy could grow significantly, with a corresponding increase in spent fuel generation rates. Like past projections that proved overly optimistic, resulting in unrealized expectations and costly missteps, the future may also disprove our current pessimistic projections relative to the future the role of nuclear energy. The nuclear industry will be well-served to acknowledge and plan for uncertainty.

4. EMBRACING UNCERTAINTY

Accommodating storage periods of unknown duration can be incorporated into design, planning, and regulatory requirements. Accepting uncertainty as a reality that must be accommodated is an empowering and overdue concept that opens new possibilities. Several initiatives have been undertaken in recent years that acknowledge and aim to address this uncertainty. A brief summary of some of these initiatives is provided below.

4.1. International Atomic Energy Agency

An IAEA consultancy tasked with considering very long term storage (e.g., greater than 100 years) considered how spent fuel storage infrastructure could be adapted to address the uncertain storage periods that will be necessary until an acceptable end point is achieved. Key concepts from their recently issued report are summarized below [13, 18]:

4.1.1. Design of future SNF storage systems

Much of the past and current effort related to extending SNF storage has focused on developing the technical basis for ensuring that existing SNF, packaging components, and related safety-significant structures, systems, and components (SSCs) will continue to perform their credited safety functions during extended storage. This is a necessary activity for extending the storage periods of existing facilities. However, because the vast majority of storage systems and facilities for the SNF that will require storage have not yet been designed or built, there are opportunities to include up-front design and functional requirements to accommodate the uncertainties of future spent fuel storage.

Future spent fuel storage (SFS) facilities and packages should consider designs that facilitate extending storage and that could adapt to different safety strategies. Adaptability will be necessary to address changing conditions, regulations, and societal values, which may occur over extended storage periods. Although these design considerations may result in increased up-front investment, the lifecycle costs may be lower than more traditional approaches that presume static conditions over the SFS facility lifetime.

SFS systems designed to accommodate extended storage must contemplate a broader range of scenarios that could occur over the longer time period. These include the potential for increased magnitude and likelihood of challenges due to natural phenomena, the accrued effects of aging, and the impacts of changing societal values and policies. Designs for extended SFS should consider increased safety margins to accommodate the potentially broader range of conditions that may be encountered during extended storage.

To ensure that safety is maintained as storage periods are extended, evaluation of the cumulative effects of both physical aging and equipment obsolescence must be an ongoing activity. Monitoring and inspection systems should consider advanced surveillance and non-destructive examination techniques to monitor storage conditions and support aging management through both preventive and predictive maintenance. Designs for SNF storage facilities, equipment, and packaging should consider the possibility that SNF, and/or its packaging, may require remediation to ensure safe post-storage transportability and/or to ensure compatibility with future SNF management steps. Hence, SFS facilities should be able to maintain, confirm, and, if needed, restore transportability.

4.1.2. SFS packaging considerations

Until an end state is identified and implemented, uncertainty will remain with respect to the optimum design for the packaging and disposal container, as well as for the acceptance criteria for the contained waste form. In the meantime, packaging alternatives should be evaluated to select a storage strategy that can be sustained over extended storage periods while maintaining flexibility and adaptability to accommodate a broad range of plausible future scenarios.

For fuel that is packaged prior to storage, a robust strategy may be to assume that repackaging will eventually be necessary and to design packaging and operational strategies accordingly. An approach that plans for periodic repackaging can (1) provide a basis for cost planning; (2) enable periodic inspection to confirm compliance with performance requirements; and (3) allow for renewal and updating of SNF packaging components and monitoring equipment. This approach also allows one to capitalize on new technologies and to address new or changed requirements. However, repackaging can add risk, cost, and personnel exposure and also generates additional radiological waste.

The alternative to packaging prior to storage is to store SNF as bare assemblies. Bare SNF can be stored in pools or in dry vaults that provide shielding and other necessary safety features. Pool or vault storage systems will require a larger initial investment and likely larger operational and maintenance costs associated with increased reliance on active systems for safety and security. However, advantages include the relative ease of access for monitoring and inspection throughout the storage period, the benefit of additional cooling time prior to packaging, the ability to capitalize on future packaging technologies and materials, and a “fresh” package for post-storage transport and handling that can be optimized for future storage concepts, criteria, and requirements. Additionally, storing bare SNF for future packaging as is done in the CLAB facility in Sweden allows repository design and selection to proceed without being constrained or influenced by decisions related to SNF packaging.

4.1.3. Regulatory considerations

Aging management and storage systems should be considered holistically by regulators, industry, and research institutions. The need for periodic license extensions should be considered and built into regulations and associated guidance to ensure that the aging management process both manages degradation and produces the information needed to demonstrate the safety of continued storage for successive licensing periods. Effective implementation of this approach can ensure compliance with requirements for as long as may be necessary.

Risk analyses provide a better understanding of the probability and consequence of specific age-related failures, which may offer insights and alternative approaches for ensuring safety. For example, some SFS safety functions are often allocated to SNF cladding, which helps to confine radiological materials and maintain the geometry of the SNF. Cladding integrity is difficult to inspect and not practical to repair or replace. For extended storage periods, safety strategies should consider shifting safety functions to packaging or facility components that can be more readily monitored and inspected and, if needed, repaired or replaced.

Performance-based approaches establish requirements based on satisfying specified performance criteria without explicitly prescribing the methods for meeting the criteria. By focusing on assuring safe conditions while leaving flexibility to the licensee as to the means of meeting established safety criteria, performance-based regulation can provide the flexibility to accommodate uncertainties such as undefined storage durations and evolving technologies and policies.

Effective use of both risk-informed and performance-based approaches will encourage development of new technologies and/or more effective approaches for addressing uncertainties associated with the need for SFS license extensions and for ensuring long term SFS safety.

4.1.4. Policy and public confidence

Delaying a final solution will result in escalating SNF inventories and management costs – making it progressively more difficult to commit resources toward reaching an end state. Delays in reaching an end state, as well as growing SNF inventories, will also negatively impact public confidence that is necessary for moving forward with lasting solutions. Therefore, a strong caution is given that policies relying on extending SNF storage, though presently necessary, be managed to ensure that commitment to achieving a sustainable solution is maintained.

4.2. U.S. Nuclear Regulatory Commission

In recognition of the uncertainty of the timing of repository availability, the U.S. NRC has prepared a generic environmental impact statement that evaluates potential environmental impacts of continued storage of commercial spent fuel over three possible timeframes: a short-term timeframe, which includes 60 years of continued storage after the end of a reactor's licensed life for operation (~100 years storage); an additional 100-year timeframe to address the potential for delay in repository availability; and a third, indefinite timeframe to address the possibility that a repository never becomes available. The results of this assessment confirm that the environmental impacts are relatively insensitive to the storage duration and that "small" impacts are possible under all three scenarios [19].

The NRC has also revised its interim staff guidance (ISG) on fuel retrievability during spent fuel storage. The former guidance, developed when a repository was expected to be operating in the near future, required that individual fuel assemblies remain retrievable during storage. Because the duration of spent fuel storage remains uncertain, the staff re-assessed the regulatory necessity of maintaining the ability to handle an individual fuel assembly. Based on the re-assessment, a new revision to ISG-2 [20] was issued which allows retrieval of SNF from storage by one or more of the following methods:

— Individual or canned spent fuel assemblies from wet or dry storage;
— A canister loaded with spent fuel assemblies from a storage cask or overpack;
— A cask loaded with spent fuel assemblies from the storage location.

NRC staff recommended that this definition of retrieval, which accommodates degradation of the SNF if safety functions are maintained, be incorporated into several other NUREGs related to spent fuel storage.

In 2016, the NRC also revised NUREG-1927 [21]. A key objective of this revisions was to expand guidance for aging management programs (AMPs) and to ensure the response to operating experience remains adequate throughout the period of extended operation (i.e., learning AMPs). The NRC has also prepared a draft NUREG-2214, "Managing Aging Processes in Storage (MAPS) Report." The MAPS Report evaluates aging mechanisms that have the potential to challenge the ability of dry storage system components to fulfil their important-to-safety functions and provides acceptable methods to identify and manage their effects. The MAPS Report also describes acceptable generic AMPs that an applicant may use to maintain the approved design basis of its storage system [22].

4.3. U.S. Department of Energy

The U.S. DOE currently manages over 200 000 SNF pieces or assemblies from various experimental, research, and production reactors that have been designed and operated over the past ~80 years. To acknowledge and address the uncertainty relative to storage durations for its SNF, the U.S. DOE is preparing a strategy that specifically considers aspects related to SNF management that may have changed since former decisions were made [23]. One of the six identified strategic goals is focused on accounting for uncertainty in the timing and availability of a path forward. A key guiding principle for developing the strategy is to maintain flexibility needed to accommodate and/or to capitalize as circumstances evolve. Specific examples include:

— Recognize the challenges and opportunities associated with the need to plan for indefinite storage;
— Acknowledge the potential impacts of evolving circumstances (e.g., new technologies, societal values, etc.) relative to SNF management strategies;
— Allow for variations in disposition pathways to account for multiple outcomes (e.g., different repository concepts, processing, continued storage, etc.);
— Avoid needlessly surrendering options.

There is a high degree of uncertainty associated with the availability dates for any large, complex industrial facility. This is particularly true with respect to nuclear facilities, given the technical challenges and institutional concerns that must be accommodated before such facilities can be constructed and operated. Identifying a disposition path is only a first step. Considerable uncertainty will remain until the end state facility, e.g., repository, and all intermediate supporting facilities and infrastructure are operating with sufficient capacity. The plan recognizes the uncertainty relative to the timing and availability of facilities and supporting infrastructure for achieving an SNF end state and incorporates that uncertainty into planning. Two specific strategies for addressing uncertainty include (1) development and use of a standard canister and (2) a plan for transitioning all SNF, except that planned for processing, to a road-ready dry storage configuration.

By relying on safety features that can be designed, engineered, and tested to current requirements rather than on SNF properties that may have large margins of uncertainty, the standard canister increases the safety and surety of operations while enabling risks to be better quantified and managed. This approach also minimizes radiological wastes and personnel exposure associated with characterization of DOE SNF. Because the sealed canister provides the credited safety features and essentially becomes the waste form, there is significant tolerance for the condition and properties of the contained SNF. This applies to both the SNF as initially placed and to its degradation behavior over indefinite storage periods. In other words, the safety case is insensitive to the condition of the SNF within the canister or its degradation mechanisms as long as they do not jeopardize the integrity of the canister itself.

Many DOE SNFs are currently stored in aging facilities in a variety of wet and dry storage configurations. In many cases, facility throughput and equipment constraints will pose challenges for transitioning to newer packaging in a timely manner. Hence, the DOE is evaluating the paths and associated needs for moving all DOE SNF to a road-ready dry storage configuration. The objective is to be able to ensure safe storage for as long as may be necessary while substantially reducing maintenance and operations costs and being prepared to capitalize on any opportunities to move the fuel to a consolidated interim storage or a disposal facility at the first opportunity.

5. INDEFINITE STORAGE – WHAT DOES IT MEAN?

Increasing storage durations and inventories for spent nuclear fuel are a reality. As illustrated above, governments are recognizing and adapting their guidance and strategies to address this reality. Industry is also responding with additional research and testing; new storage concepts, equipment, and facilities [24–26], as well as guidance and support for safely extending storage within existing equipment and facilities [27]. However, the necessary technical and regulatory solutions are unlikely to be enough to clear the path for nuclear to play a vital role as a large-scale, low-carbon energy source[5]. Discrimination against nuclear as a low-carbon energy source is not rooted in technical issues. Rather, it is primarily rooted in public attitudes, which translate into discriminatory public policies [16]. As with past efforts to effectively manage and dispose of spent fuel, social and political opposition, not technical issues, will likely continue to be the key obstacle.

Failure to achieve sufficient public acceptance has been a persistent source of difficulties, delays, and challenges to maintaining the political will needed for successful siting and licensing not only a repository for final disposal of SNF and HLW but also for other facilities needed for effective spent fuel storage and management. This is a key factor in missed commitments and continued difficulty making substantive progress, which reinforces risk perceptions and further erodes public confidence that the nuclear waste problem can be solved. This, in turn, increases both the quantity of SNF and the duration of SFS, while further increasing the challenges associated with siting and licensing SFS facilities.

This circular effect is illustrated in the lower loop of the figure below, which shows that a lack of public confidence and political will to address the problem can complete a feedback loop that aggravates the original problem – further impeding a lasting solution. However, it should be noted that public confidence is influenced not only by perceived risk but also by perceived benefits – suggesting that both areas present opportunities for bolstering public confidence.

A key observation from the figure below is that public confidence is actually influenced by the perception of both risks and benefits. Hence, although adequate safety performance is necessary, reducing risks and risk perceptions alone is unlikely to be sufficient to reverse the cycle. Benefits must also be recognized and valued. Building the public confidence requires that the public recognize the opportunity for real benefit from nuclear, and/or to avoid real cost, relative to their values. Absent this incentive, there is no motive for constituents to reconsider their risk perceptions [17].

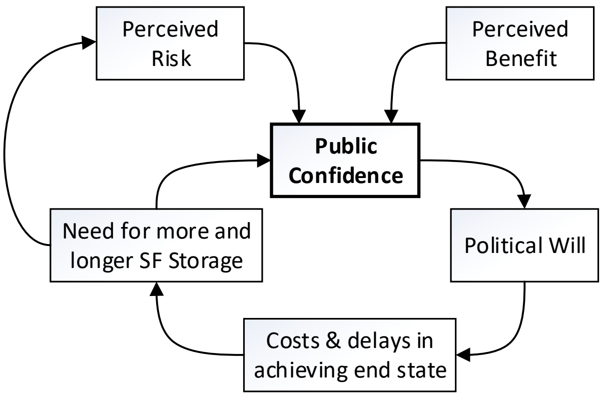

FIG. 3. The public confidence dilemma – or opportunity [18].

So, although we cannot credibly argue that we know how long SNF must remain in storage, we can make a credible case that SNF can be stored safely for as long as may be necessary, that nuclear energy can provide significant environmental and economic benefit in a carbon-constrained world, and that it can be managed safely and sustainably without passing an unacceptable burden to future generations. The spent fuel management community must do its part. In addition to driving towards an acceptable solution for completion of the nuclear

[5] Though many consider nuclear to be a carbon-free energy source, fuel production, plant construction and operation, and decommissioning and disposal of the plant and its spent fuel are not carbon-free. Similarly, solar, wind, hydroelectric, or any other power source is not carbon-free. Use of the term low-carbon here is intended to encourage consideration of accurate life-cycle carbon footprints when comparing energy sources.

fuel cycle, we must face the current reality of indefinite storage — meaning 1) that spent fuel storage durations and inventories will continue to increase for the foreseeable future, 2) that the future is not foreseeable, 3) that this uncertainty need not and should not pose a problem, and 4) that our industry infrastructure, communications, and policies can and must look ahead, confront, and openly address the unknown durations and increasing inventories.

6. EPILOGUE

This paper is not intended, in any way, to discourage commitment to achieving a sustainable end state[6]. It does advocate for accepting the reality that SNF storage periods cannot be credibly defined and for designing and planning our infrastructure accordingly.

Reviewers have expressed discomfort with the idea of storing SNF over undefined periods. An IAEA Consultancy encountered similar concerns from reviewers of the IAEA Technical Report entitled 'Storage of Spent Nuclear Fuel until Transport for Processing or Disposal' [18]. The source of this discomfort seems to result primarily from two issues.

- A perceived conflict with the desire to maintain pressure on the policy makers to follow through with implementing geologic disposal;
- Concerns relative to the need to maintain institutional controls over extended periods along with the potential of passing obligations to successive generations.

Both of these policy considerations are valid and should be addressed openly and directly. SNF can be stored until another pathway becomes available without stoking concerns that safety may be unacceptably compromised in the interim. We simply cannot predict when a better solution will become available. But we can, and must, ensure safe storage until then. By openly acknowledging and accepting this, our design, operational, and regulatory frameworks can incorporate the features needed to ensure safe and effective storage in the interim -- as well as a smooth transition to a subsequent phase or the final end state.

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Paper ID#161
COST/RISK-OPTIMISED FUEL CYCLE DECISION-MAKING IN UNCERTAIN MARKET FUTURES

L. VAN DEN DURPEL
Nuclear-21
vddurpel@nuclear-21.net

Abstract

The various options for the management of spent fuel (SF) from nuclear power reactors is a topic that has been debated from multiple dimensions, being it the socio-political concerns with regard to geological disposal, the technical-economic competitiveness of options such as reprocessing and recycling, as well as from the growing discussion on sustainability and international policy.

Many of the discussions relating to spent fuel management have historically been rather binomial between, on the one hand, the socio-political concerns on the direct disposal of spent fuel and the proliferation concerns regarding reprocessing, and, on the other hand, the uncertain costs of such disposal facilities versus the economics of reprocessing and recycling schemes. Especially since the 1990s, various intergovernmental and national organizations-initiated studies on very advanced spent fuel management schemes such as separation and transmutation also impacting the progress towards a proper solution-oriented and responsible and above-all timely spent fuel management.

After some decades of - generally - indecisiveness on spent fuel management, and with nuclear energy increasingly in the spotlight in the context of sustainable energy mixes, a more solution-oriented and responsible spent fuel management becomes necessary, if not urgent.

Especially as the uncertain costs and timing for such spent fuel management become increasingly translated into financial risks for the spent fuel owners, i.e. utilities. Many discussions on spent fuel management options were in the past colored by strategic reflections on natural uranium availability and pricing, sustainable nuclear fuel cycle options (including Generation-IV systems [1]) and political considerations regarding non-proliferation. Today, there is a growing financial risk presented to utilities which becomes a more compelling trigger towards a decision on various spent fuel management options.

This paper addresses the changing market context for nuclear energy and particularly how spent fuel management options are increasingly assessed in such uncertain futures. Cost/risk optimizing spent fuel management schemes are crucially important for utilities not to have spent fuel as such remaining a hurdle for the future of nuclear energy's use.

1. CONTEXT

The deployment of the nuclear power plant park (NPP) worldwide goes with the continuously increasing amount of spent fuel to be managed as pictured in Fig. 1 and Fig. 2 [2]. Figure 1 pictures the time-evolving NPP park since mid-last century and projecting the NPP-park as currently operating and under construction. Figure 2 shows the resulting spent fuel inventory under a business-as-usual scenario (i.e. without modification of today's fuel cycle option). Under such business-as-usual scenario, a doubling of the world's spent fuel inventory is to be expected during the next 25 years with potentially even a larger growth of spent fuel inventories assuming additional new build NPPs as increasingly projected in decarbonizing energy mix scenarios.

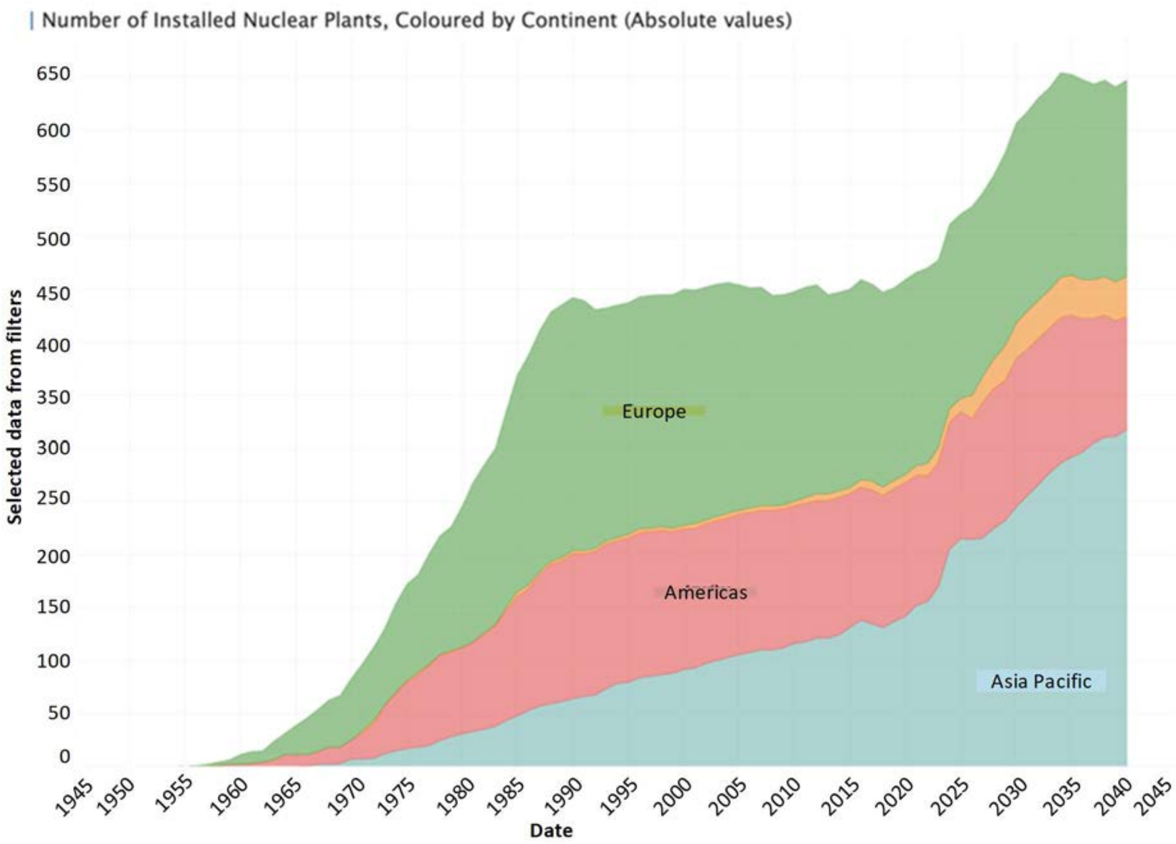

FIG. 1. World's NPP-fleet evolution since mid-last century.

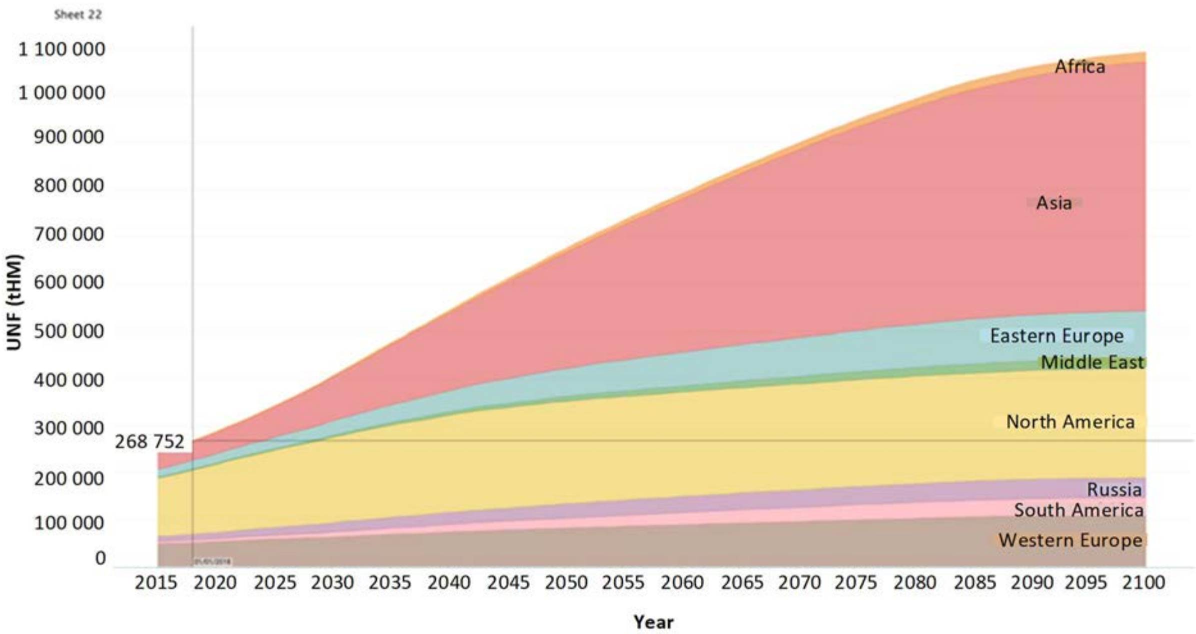

FIG. 2. SF-inventory in Business-as-Usual scenario for the current operating and under-construction NPP-fleet.

Awaiting decisions on the final destination of this spent fuel, most of the spent fuel remains interim stored in dry interim storage solutions at reactor sites with the remainder awaiting in at-reactor pools for cooling before being transferred to interim storage. Reprocessing of spent fuel being practiced since the 1970s has resulted into the reduction of the SF-inventory in many countries and for some 30% by now of the spent fuel inventory worldwide.

The deployment of final geological disposal repositories (GDF), the prime solution for the ultimate management of high level waste, hasn't yet been following this trend as no GDF for commercial NPP spent fuel is operational today [3]. And even if so, the pace of GDF deployment will probably not match the SF-inventory growth during this century as the first GDFs to become operational foresee accepting conditioned SF from the late 2030s on. Many other countries projecting operational GDFs only from well into the second half of this century or just from the 22nd century on.

The option to reprocess SF and recycle the reprocessed uranium (RepU) and plutonium (Pu) is practiced by some countries since the 1970s and has been industrialized by France, UK, Russia and soon Japan, India and China. The recycled fuel as MOX-fuel and REPU-fuel being a mature fuel option for light water reactors.

So-called 'Generation-IV' nuclear energy systems have been presented during the last 20 years without a real industrial deployment of such systems expected before mid-century. The prime Generation-IV NPP-type being sodium-cooled fast reactor (FR) has been designed and operated by France, United Kingdom and Russia and continues so by Russia with, soon to be, also India and China and possibly US. The reprocessing of the LWR-origin SF being anyhow a central requirement by any of these Generation-IV nuclear energy systems and even more so for advanced nuclear energy systems that have been researched for the last decades, e.g. molten salt reactors (MSR) and accelerator-driven systems (ADS).

A central challenge remains how such more advanced nuclear energy systems developed within essentially a governmental-strategic approach may be fitted within an economic driven energy market future? And this especially for economic decision-making which span multiple decennia, and which is typically way beyond a utility's decisional horizon? The answer to these questions relates to financial risk management as the rest of this paper will document.

2. WHAT ARE THE TANGIBLE OPTIONS FOR SF-MANAGEMENT BY MID-CENTURY?

With nuclear energy potentially becoming a substantial contributor to sustainable energy mix futures worldwide, the question arises which SF-management options are truly tangible solutions today and within the coming decades?

While the growing spent fuel inventory worldwide is pictured in Fig. 2, Fig. 3 shows this same spent fuel inventory from the perspective of age of the growing dry interim stored spent fuel. Ageing interim stored spent fuel is one of the issues to be addressed given the constantly delayed GDF deployment calendar. Interim storage of spent fuel is a safe and economically attractive option awaiting future reprocessing or disposal of the spent fuel though the potential degradation of the, especially dry, interim stored spent fuel may be very problematic if this SF needs to be reconditioned after (very) long periods of interim storage. As much of this interim stored spent fuel still resides on NPP sites, with possibly already many sites without operational NPP, this reconditioning can be a very substantial cost exposure. Dedicated hot cell construction may be required to recondition into, at least, dual purpose containers for continued interim storage or transport to other storage or reprocessing or disposal facilities.

Figure 3 shows the spent fuel inventory for only the current operational and under-construction NPPs the amount of interim stored spent fuel indicating the 'frontier' of 60 and 80 years of interim stored spent fuel. There will be a significant amount of spent fuel reaching the 60-years interim storage duration by mid-century with an increasingly important amount of spent fuel that has been stored for more than 80 years from mid-century on. According international studies on ageing dry interim stored spent fuel, significant investments may be expected to recondition this ageing spent fuel especially beyond the 80 years interim storage time as most of this spent fuel may be still residing on shut-down NPP sites and thus requiring a transfer to centralized interim storage sites or sites allowing further processing of this spent fuel, being it for disposal or for reprocessing.

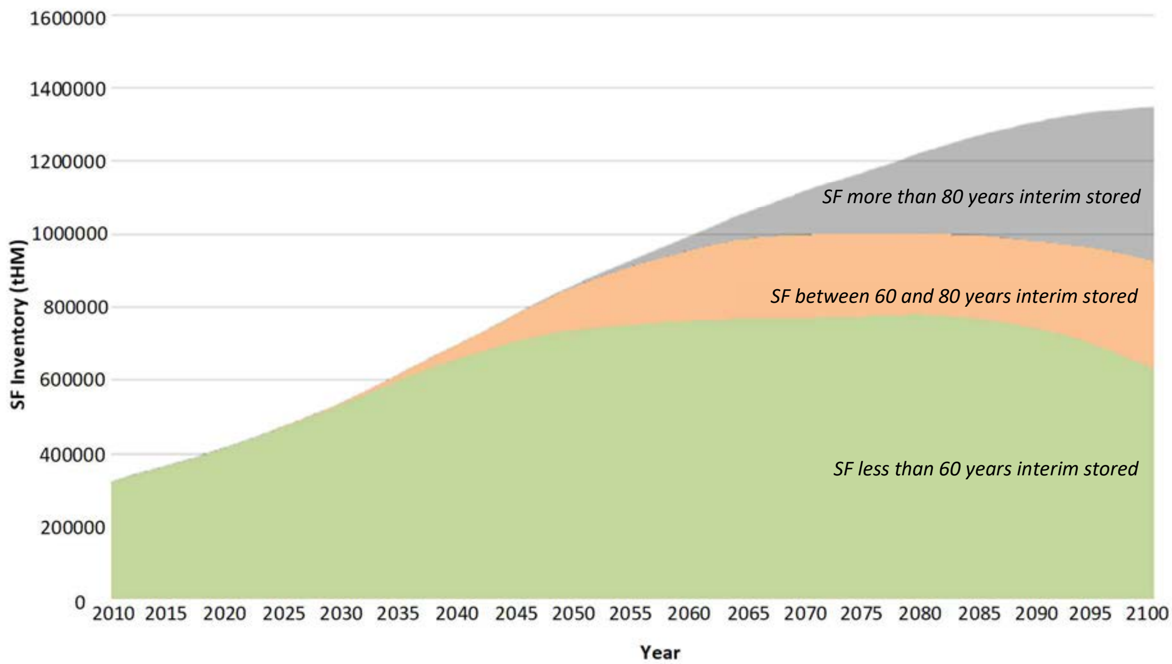

FIG. 3. Evolution of SF: Inventory for the foreseeable NPP-fleet with distinction for interim storage duration of SF.

Some countries having such aged spent fuel will not yet have geological disposal sites in operation when an increasing spent fuel amount reaches the 80-years frontier with thus an increasing technological and above-all financial risk arising.

A variety of other spent fuel management options are or may become industrially available during the coming decades (Fig. 4), i.e.

— Reprocessing of UOX-fuels and recycling of uranium and plutonium is an industrial practice in LWRs today with progress being made allowing for additional reprocessing and recycling of the MOX-fuel with multi-recycling of Pu in (TOP)MOX, CORAIL, MIX or REMIX fuels;
— So-called Generation-IV nuclear energy systems essentially using fast reactors (FR) may become industrially available by mid-century with some demonstration plants currently under operation, construction or consideration in Russia, India, China and possibly France and USA later-on;
  - Though, the pace of deployment of such FRs will not match the possible urgency to manage the aged spent fuel inventory from mid-century on;
  - HTGRs may deploy earlier as part of more sustainable energy mix policies including the use of HTGRs for non-electric applications. Such HTGRs may serve, in parallel, a Pu-burning mission contributing to the SF-management essentially from LWRs;
  - Molten Salt Reactors (MSRs) are currently again subject if increased interest though industrial deployment is not to be expected well before 2050. Even if their deployment would come significantly earlier, the deployment of associated fuel cycle services will not signify a real contribution to LWR's spent fuel management soon.

Finally, very advanced options as Accelerator-Driven Systems (ADS) are to be projected well into the second half of this century, if ever required to be realised, in furthering closed nuclear energy systems. Their prime motivation being the transmutation of minor actinides being an option beyond the management of Pu. Such Pu-management remains the prime objective with secondly the management of reprocessed uranium amounts towards a true SF-management before such minor actinide management could further improve waste management.

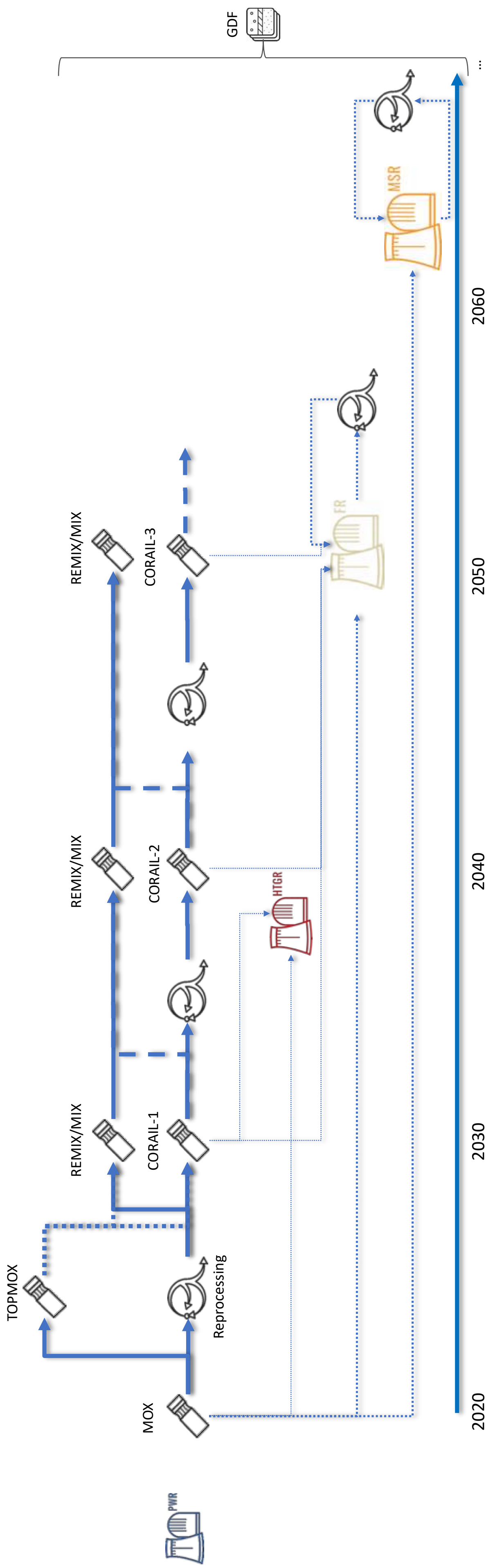

FIG. 4. Projected deployment of nuclear energy systems towards more advances SF-management.

The next two decades will be crucial to demonstrate the proper deployment of ultimate SF management options beyond the continued (dry) interim storage of this spent fuel.

3. THE DRIVERS FOR SF-MANAGEMENT DECISION-MAKING BEING THE COST/RISK EXPOSURE FOR THE VARIOUS STAKEHOLDERS

Past discussions on SF-management and particularly the choice between direct disposal and reprocessing/recycling were mostly governed by two prime considerations:

— Since the early days of nuclear energy use, the availability of deemed scarce natural uranium resources were considered as a clear driver towards closed fuel cycles minimising the amount of natural uranium to be mined. The introduction of FRs and, as intermediate step, LWR-MOX were considered prime options towards such reduction in natural uranium requirements while also reducing the amount of SF;
— Since the early 1990s, given the continuous delay in geological disposal facilities deployment, the ‘clean waste, dirty fuel’ idea got traction in various countries seeking to minimise the amount and radiotoxicity of resulting waste to be disposed of. The thought being to reduce this amount and radiotoxicity as being considered the prime socio-political objections against nuclear energy. More than 2 decades of significant R&D hasn’t yet resulted in progress towards the industrial deployment of such very advanced nuclear energy systems and multiple decades are still required before this may become reality.

During this period, the SF-owner i.e. utilities hands-in-hand with fuel cycle service companies have continued to further evolutionary progress towards the safe and economic management of SF by deployment of interim SF storage solutions. However, such interim SF storage solutions are not ultimate SF management solutions and need, in due time, to transition towards such final solutions.

Today, the situation on SF-management is changing with the cost and financial risks for utilities as SF-owners being the prime drivers in decision-making on SF, e.g.:

— GDF-programmes remain delayed with increasing needs for utilities to provision for ever growing budgetary projections for GDF-programmes;
— Utilities may need to provision as well for extended (dry) interim storage even beyond the operational lifetime of the NPPs;
— In addition to the transfer of the spent fuel from such ‘sunk’ NPP-sites towards centralised interim storage options awaiting the final destination for the SF;
— Changing regulatory framework may add to a changing context and particularly economics of longer-term interim storage options, and
— Many other influencing factors as there are national programmes, safety/security and safeguards, etc.

The decisional framework is not as such the natural uranium availability or any very far future radiological risk, but the financial risks associated to uncertain spent fuel management deployment. Cost/risk-reducing scenarios for spent fuel management become hereby central in the decision-making on spent fuel as pictured in Fig. 5.

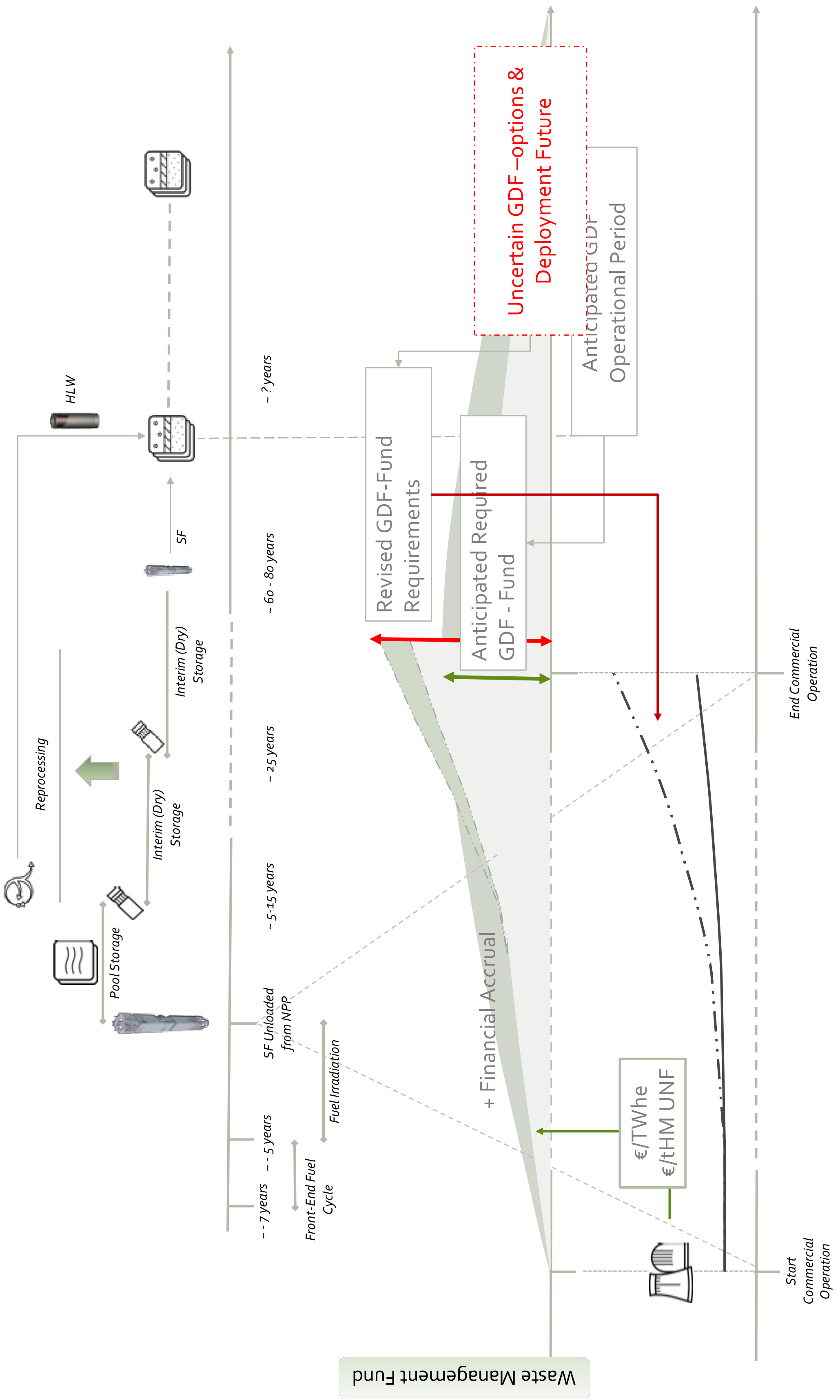

FIG. 5. Financial risks from uncertain SF-management futures.

An increasing number of utilities are being faced with increasing waste provision tariffications due to the uncertain timing and costing of spent fuel management programmes particularly the GDF-deployment. Especially for utilities with eldering NPP fleet, i.e. beyond half of the expected lifetime of the NPPs, these higher tariffs for GDF may be impacting the market rating and overall economic competitiveness of the NPPs and NPP owner.

4. COST/RISK REDUCING SCENARIOS FOR SF-MANAGEMENT

A cost/risk reducing decisional framework is therefore increasingly central in utilities' SF-management decision-making. Such a cost/risk-reducing decisional framework seeks:

- To assess the expected and uncertainty distributions for the future costs related to various spent fuel management options;
- To define and assess the decisional moments when one may switch between spent fuel management options;
- To value the decisional flexibility at these (also uncertain) decisional moments;
- Allowing to design decisional scenarios minimising costs and risks over time such that the overall financial cost/risk exposure from spent fuel management is minimised and matched to the stakeholder/utility's financial risk appetite.

Such a cost/risk reducing decisional framework model (i.e. NROM developed by Nuclear-21 as part of the NESSAT toolbox) on spent fuel management is demonstrated in what follows on a generic spent fuel management case where spent fuel management options of direct disposal, partial recycling with MOX-fuel and closed fuel cycles with FRs was considered for a large and eldering NPP-fleet with significant amount of spent fuel in dry interim storage.

Figure 6 summarizes the various decisional options a utility may take starting from the management of spent UOX-fuel. A utility may decide to extend the pool interim storage or the dry interim storage awaiting progress in GDF or may decide to switch to early or delayed reprocessing during the interim storage of spent fuel. The reprocessed materials as reprocessed uranium and plutonium may be recycled once or multi-recycled as (TOP)MOX/CORAIL/REMIX/MIX-fuel.

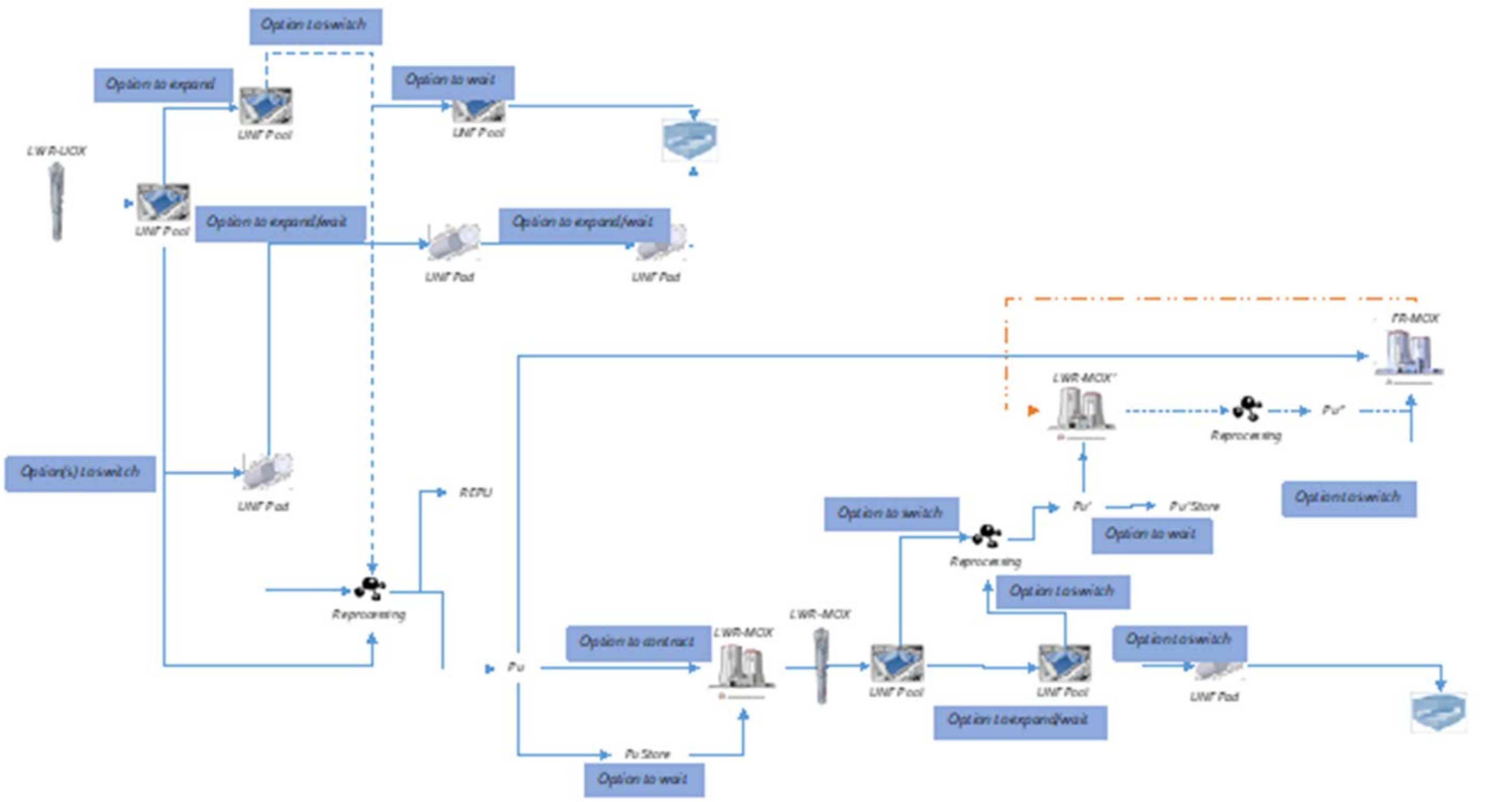

FIG. 6. Schematics of decisional options in SF-management

Such a more complex decisional framework covering multiple decades as well as many stakeholders having influence on the decisional options and timing of decisions can be analyzed with appropriate risk assessment methodologies such as real options analysis.

Applying such methodology, i.e. NROM Nuclear Real Options Model, Fig. 7 generalizes the outcome of such assessment with Fig. 8 and Fig. 9 picturing the results of a real (though generalized) case example.

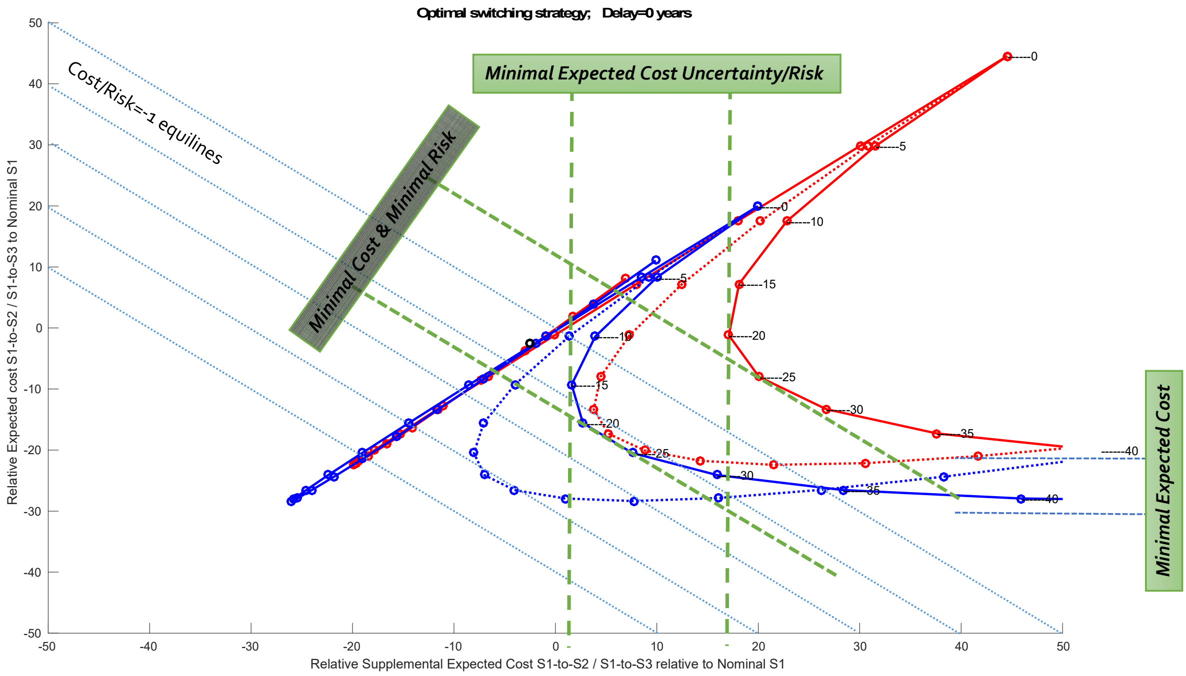

FIG. 7. Cost/risk decisional framework on SF-management comparing direct disposal and partial recycling option.

Figure 7 pictures the relative cost (vertical axis) with the risk for relative additional cost exposure (horizontal axis) for a partial (MOX) recycling scenario compared to the direct disposal route and this for two cases depending on historic/nominal (red) GDF-costing and updated higher (blue) GDF-costing. At time = 0 (upper right), the equivalent NPV-analysis indicates that there's no immediate interest to perform partial recycling though, as time progresses and uncertainties from SF interim storage performance and the need for expensive reconditioning and uncertain GDF timing and costing become more apparent or closer-by, partial recycling may lead to a significant reduction in both cost and risk exposure as shown around t = 15 – 25. This would indicate that the optimal time to execute partial recycling strategy would be some 2 decades into the future for this case as it would then minimize both cost and risk exposure compared to the direct disposal route. The dotted variants of the two curves show the impact from larger uncertainty distributions on the spent fuel interim storage and GDF costing and timing. Such cost/risk-optimized scenarios therefore do not lead to a classis 'yes/no'-decision but values the time towards decision during which changing exposure to uncertainties may develop. This also allows to mitigate such future options and to value the mitigation options matching a stakeholder/utility's cost/risk-appetite and financial performance.

A real case though generalized example in Fig. 8 and Fig. 9 shows how such a cost/risk assessment allows to perform portfolio analysis on the amount of spent fuel interim stored to decide when which amount of spent fuel would be ideally switched to other then direct disposal spent fuel management routes. Figure 8 summarizes a portfolio analysis for an amount of SF in a country covering both lower burnup (BU) longer-cooled SF (UNF2), intermediate BU and dry stored spent fuel (UNF4) up to higher BU pool stored spent fuel (UNF5). Where typical assessments of spent fuel management options address the spent fuel inventory as a whole, this cost/risk portfolio analysis investigates the various spent fuel types and their corresponding cost/risk-exposure allowing to:

— Segment the spent fuel inventory according the cost/risk-exposure and prioritisation for spent fuel management decision-making;
— Minimise the overall cost/risk-exposure over time by optimising the investment and operational costs while optimising the hedging or mitigation of future financial risks.

This example in Fig. 8 indicates that the UNF1 spent fuel wouldn't rank for reprocessing scenarios where the UNF4 would need prompt decision towards reprocessing while the UNF5 spent fuel inventory remains attractive for reprocessing scenarios until some 35 years into the future though with minimal cost/risk exposure in 15-15 years, i.e. deciding now. The expected cost savings from optimal switching for the different spent fuel inventories being shown in the lower part of Fig. 8 indicating a potential expected cost saving of about 25 B$ by 2050 for the considered scenarios of switching from direct disposal towards reprocessing scenarios for some of the spent fuel inventory (UNF4 and UNF5) while other destined for geological disposal (UNF2).

Given that UNF4 represents a significant fraction of the total spent fuel inventory, a sensitivity analysis of decisional factors is presented in Fig. 9, i.e. the relative cost (vertical axis) and relative risk (horizontal axis) for various SF-management routes for the UNF4-category of spent fuel are pictured for:
— S2 = partial LWR-MOX recycling scheme;
— S3 = closed fuel cycle scenario with transition of LWR towards FR;
— S4 = symbiotic LWR-UOX + LWR-MOX + FR scenario mitigating the development risks for FR with mature technology such as partial recycling using MOX.

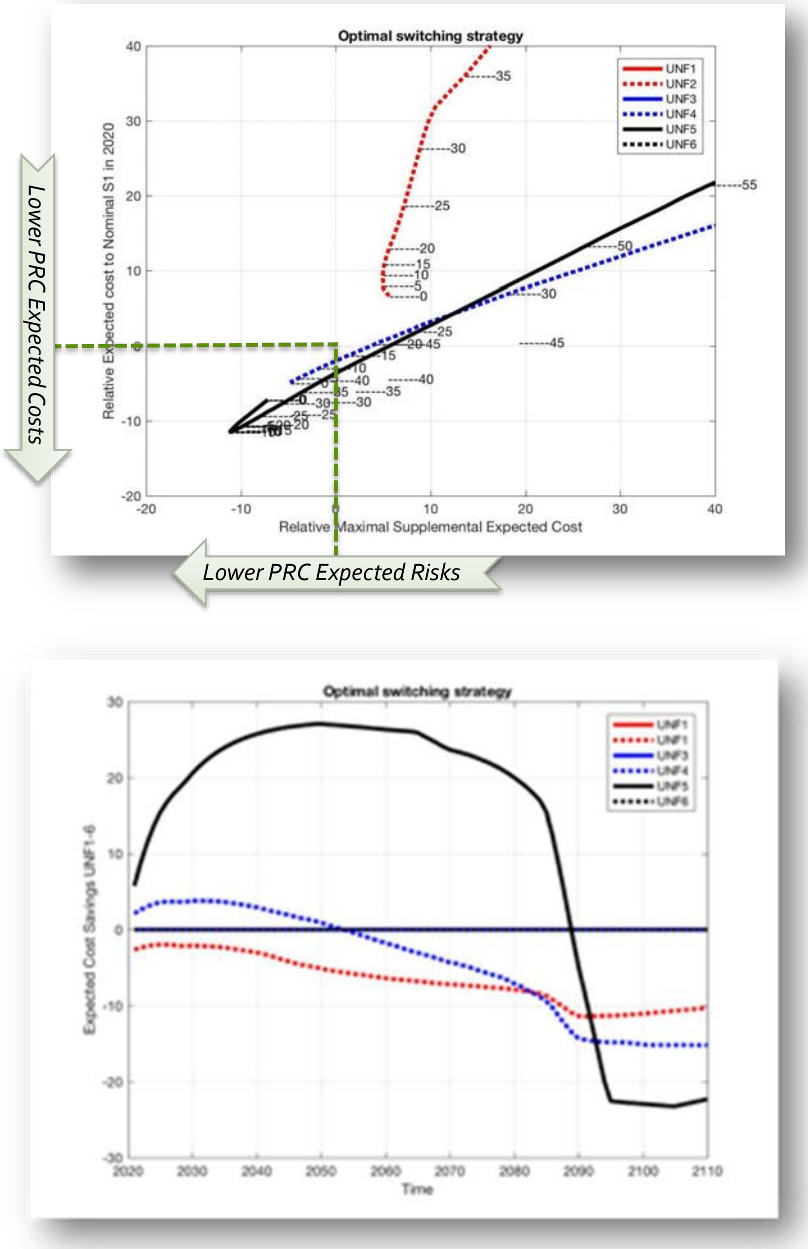

FIG. 8. Portfolio analysis of a country's SF inventory.

For various assumptions on GDF-costing and timing and uncertainty, as well as for FR cost/timing uncertainty, the optimal spent fuel management schemes differ significantly between the SF-management options and the ideal timing of execution of these options. In this real case example, partial recycling options were recommendable by around 2035 where fully closed fuel cycles following LWR-UOX use did not rank as optimal given the high cost and risks from insufficient FR maturity not compensating yet for the GDF cost/timing uncertainties.

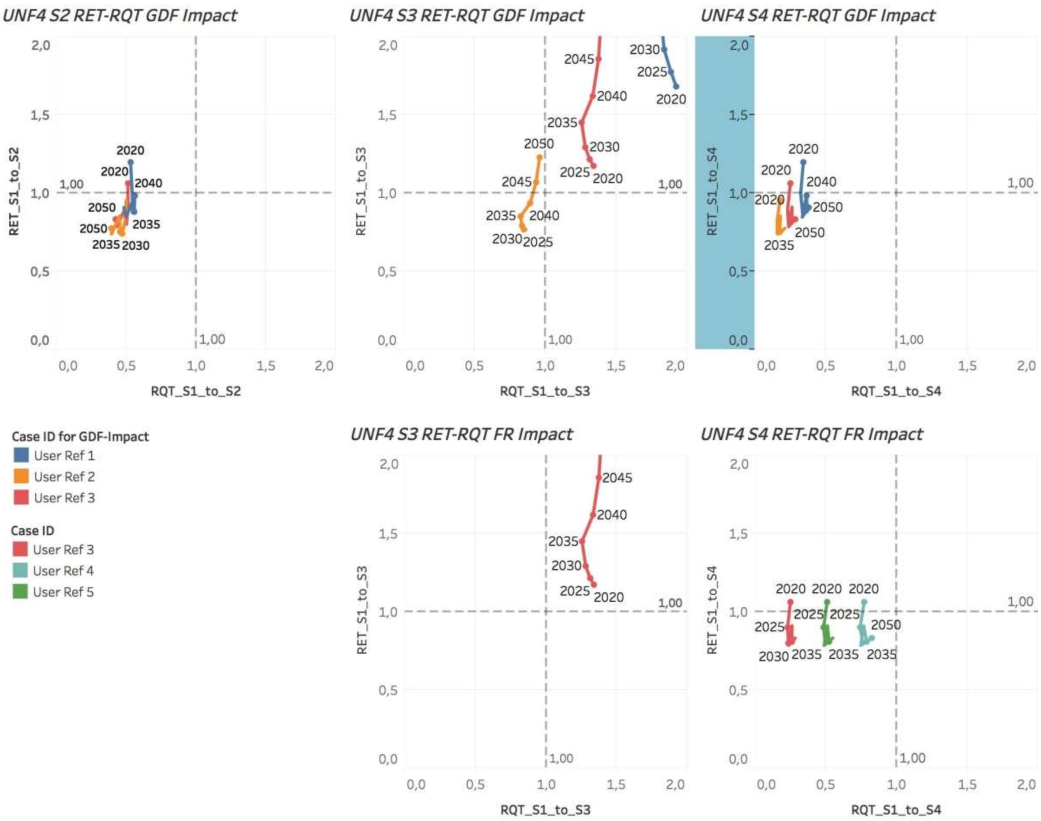

FIG. 9. Generalized though real case application of cost/risk optimized decision-making on SF-management.

5. CONCLUSIONS

Decision-making on SF-management doesn't relate exclusively on strategic reflections on natural uranium availability and radiological risk reductions in the long term but increasingly on the cost/risk exposure for the spent fuel owner, i.e. utilities. The decision-making is now geared towards the reduction of cost and financial risks in uncertain energy markets and still uncertain timing and costing for GDF deployment next to uncertainties occurring in long term interim storage of spent fuel. Other uncertainties relating to the development roadmap for more advanced nuclear energy system options where the past 20 years have shown a rather continuous delay on development and deployment.

Cost/risk-decisioneering methodologies allow to assess the optimal timing when which spent fuel management options could be executed minimizing the cost and risk exposure of utilities to spent fuel management. Such methodologies allow, among others:

— Spent fuel owners/utilities to assess their cost/risk exposure from spent fuel management;
— To support decision-making on SF-management and providing a more informative decisional framework on when what option to consider for investment/execution;
— Portfolio management;
— Fuel service company development and commercial offer analysis;
— R&D investment analysis on cost/risk-reduction impact.

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3.8. YOUNG GENERATION CHALLENGE

The #SFM19 Young Generation Event (YGE) Challenge was conceived to increase participation in the conference by Young Professionals and Students. The Challenge call encouraged the submission of papers based on the seven Tracks of the Conference, and also welcomed submissions relating to the conference scope. Thirty-five eligible abstracts were received, each of which was accepted into the conference by the ISPC as either an oral or poster presentation. An IAEA internal panel, comprising of representatives from the two divisions organizing the conference, selected four winning entries. The four winners each received a travel grant to attend the conference, and the opportunity to present their paper and co-chair a session in their selected track. At the end of the conference, the audience were asked to choose the overall 'Winner of Winners', revealed during a prize giving ceremony where each received a certificate signed by the DDGs of the NE and NS&S Departments.

The winning entries received were:

- **Paper ID#47 by B. Ficker (Hungary)** Storage capacity enhancement of SFISF at Paks in Hungary
- **Paper ID#10 by A. Kirkin (Russian Federation)** Approaches to evaluation of spent nuclear fuel reprocessing products activity and volume equivalence which is returned to a supplier state in the Russian Federation
- **Paper ID#34 by T. Okamura (Japan)** Reduction of geological disposal area by introducing partitioning technologies under conditions of high burnup operation and high content vitrified wastes (selected as the Winner of Winners)
- **Paper ID#54 by J. Home (United Kingdom)** Strategies for post-closure long term information management

Paper ID#47
STORAGE CAPACITY ENHANCEMENT OF SFISF AT PAKS IN HUNGARY

B. FICKER, B. GÉHER
SOM System Ltd.
Budaörs, Hungary
Email: balazs.ficker@somsystem.hu

Abstract

Spent fuels (SF) assemblies from Paks Nuclear Power Plant (Paks NPP, Hungary) are placed in Spent Fuel Interim Storage Facility (SFISF) since 1997. The SFISF is a modular vault dry storage (MVDS) type design accommodating SF after a minimum of a few years of cooling time in the reactor decay pool. The SFs are stored individually and separately in the vault modules (VM) in airtight sealed fuel storage tubes (FST) filled with inert gas. Decay heat rejection is achieved by buoyancy driven air flow through the vault, passing over the exterior of the array of storage tubes.

The capacity of the SFISF was planned on the total amount of the SFs arising from the planned 30-year lifetime of Paks NPP. To store these SFs a 33 vault facility was designed with 450 FST in each vault. Until now all together 24 vaults have been constructed.

Sixteen vaults were built with 450 FST in each vault. To make the storage economically more efficient the number of FSTs was increased from 450 to 527 in the last eight vaults. This was provided by use of the built-in reserves of the design and the development of analyses techniques making it possible to reduce the conservatism in calculations. According to this modification the total capacity of the SFISF was increased by around 9%.

At the millennium a decision was made to extend the lifetime of the Paks NPP with addition 20 years, resulting a significant growth in the amount of the SFs. In order to adjust the storage capacity a review of the design was carried out. The structural analysis showed that a number of 703 FSTs could be installed into the same geometry by modifying the charge face structure (CFS). Based on this number the total capacity could be increased by almost 20% compared to the original design.

Considering the initial few years of cooling period and applying it for the whole storage facility the heat load could be higher than the design criteria. However, with the rearrangement of the SFs cooled for many years in the FSTs it is possible to solve this issue. The decay heat production of SFs stored for many years decreased to a level at which it is possible for them to be placed in a higher density redesigned vault with the new CFS design. By transferring the older SFs to the higher density vaults there will be enough free positions to place the newer SFs arriving from the NPP. Construction license with the newly increased storage arrangement was issued by the nuclear authority in 2017.

The paper describes the design, modelling and licensing process of this capacity enhancement.

1. INTRODUCTION

1.1. Selection of the storage facility

According to the fuel strategy that was effective at the time of construction of the Paks NPP the Soviet Union undertook to take back the SF for reprocessing without returning any product or waste from it. The first transport of SF took place in 1989, but altogether 2331 SFs were returned.

As a result of a selection process the Modular Vault Dry Storage system was selected from a group of equally safe and reliable storage technologies in the beginning of the 1990s. The main factor of this decision was the fact that the MVDS technology has provided the lowest SF cladding temperature during storage. Having only limited experience at that time on the behavior of the VVER-440 type SF under dry conditions it was judged to be an important issue. It was the reason that the operator of the Paks NPP signed a contract with a British-French company GEC Alsthom Engineering Systems Ltd. to build a dry storage facility of MVDS type. By 1997, the first VM — containing three vaults — and the service building has been built.

The possibility of the use of Russian reprocessing services still exists, but since commissioning of the Paks storage facility all SF assemblies taken out from the decay pools of the reactors are stored at SFISF adjacent to the NPP.

1.2. Facility design

The Paks storage facility functionally can be divided into three major structural units (Fig. 1).

The first major unit is the service building in which the reception, preparation, unloading and loading of the transfer cask takes place. The fuel assemblies are transported to the MVDS from the at-reactor pool using the C-30 transfer cask (with a maximum capacity of 30 FAs) and its railway wagon. The fuel handling system and other auxiliary systems are installed in this building.

The second major structural unit is known as the charge hall where the fuel handling machine travels during the fuel handling operations. The charge hall is bordered by the reinforced concrete wall of the ventilation stack on the one side and by a steel structure with steel plate sheeting on the other side.

The third one is the VM where the SF assemblies are stored in the vertical tubes (Fig. 2). These VMs include a minimum of three or maximum five vaults depending on the geometrical arrangement.

The VM structures form a rigid enclosing ‘box’ with substantial thicknesses of radiological shielding concrete, which also provide adequate structural strength and weather protection. The box cell structure (i.e. the vault module) is supported by an integral foundation raft bearing directly onto the replacement fill.

The outlet ducts form stiff vertical cantilevers from the cellular structure, with thicknesses determined largely by shielding requirements. The rigid concrete structure provides firm anchorage points for the steelwork forming the charge hall enclosure. The steelwork is adequately braced in the plane of the walls and roof to ensure the elimination of sway and to bring the reactions directly on to the concrete structures.

Cooling air enters the vault through a louvred opening which is provided with a mesh covering to prevent the ingress of birds or large debris. The individual inlet openings are connected to a common plenum to aid the vault airflow distribution and to maintain the flow if an opening becomes blocked with snow or other debris. The air passes through a concrete labyrinth, which provides radiological shielding of the fuel assemblies then into the vault tube array section via precast concrete collimators, which are cast into the main cell structure walls. The collimators provide further radiological shielding of the fuel assemblies, whilst improving the cooling air distribution through the vault.

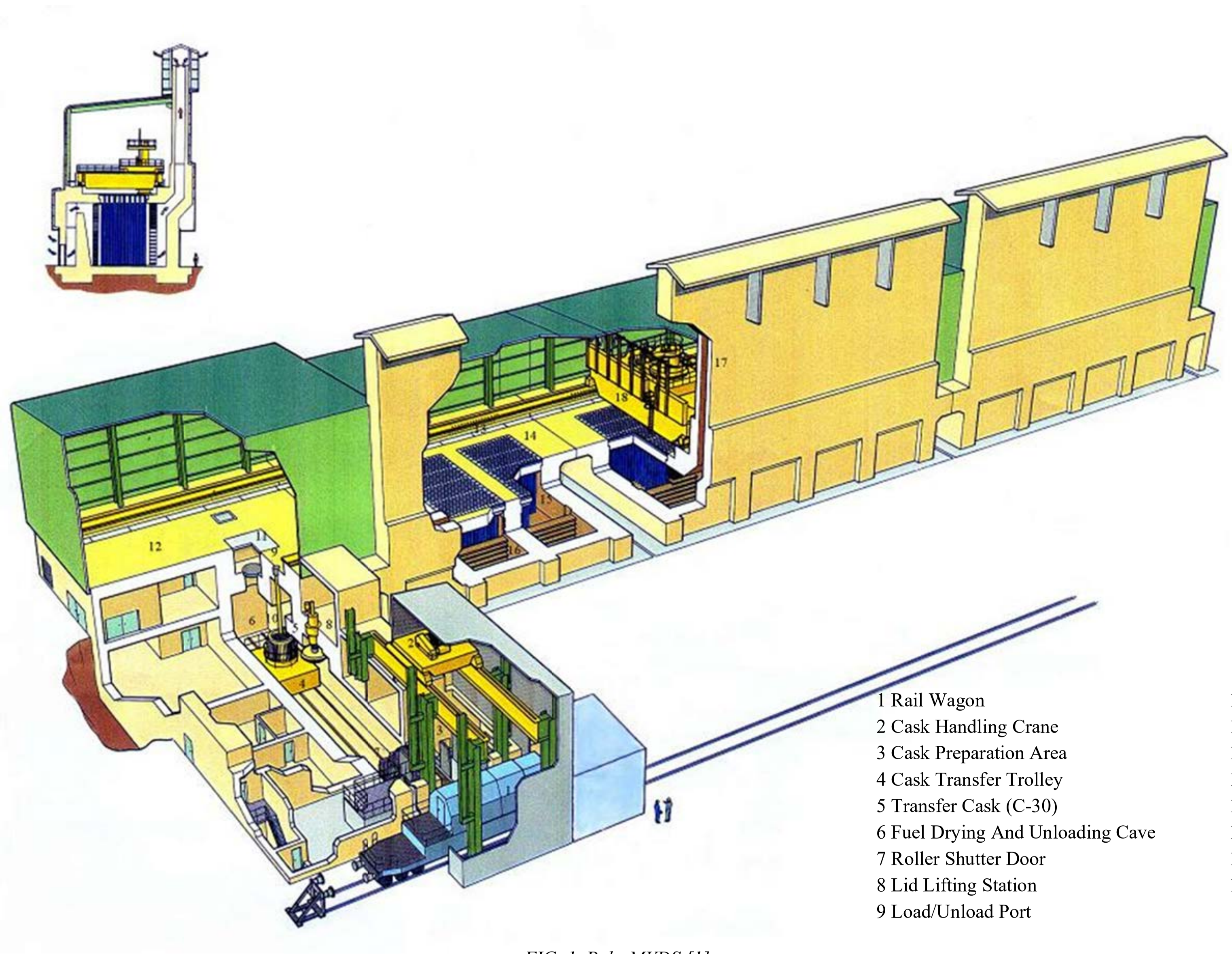
The air leaves the vault through a second set of collimators and is exhausted to the atmosphere through a concrete outlet duct which does not come in contact with the fuel in their storage tubes. Thus, the internal surfaces of the vault will remain clean and will not require decontaminable finishes.

The vault floor provides support to the FSTs via grouted-in support plates. A grouted gap provided in the top of the vault walls supports the CFS. Each CFS consists of four pre-fabricated steel boxes filled with concrete at site for shielding purposes. The CFS forms the roof of the vault and provides horizontal support for the FST array. The vertical loads are transmitted to the civil structure in direct bearing [1].

2. CAPACITY NEEDS

Due to its modular nature the MVDS facility has been constructed according to the operational needs of the of the NPP. Initially the operator of the Paks NPP specified two requirements regarding its SF storage capacity needs. One of them was to accommodate the SF amount generated by the four reactors of the NPP in 10 years operation. The other one was to make it possible to extend the facility to receive the additional remaining SFs generated through its originally designed 30 year service life. The latter requirement has particular importance since parameters which depend on the number of SFs had to be taken into account for the full deployment of the facility. As a consequence, parameters which concern for example radiation protection had to be justified for the case of storing all SFs. According to the first construction license an 11 vault facility was erected as it is shown in Fig. 1. with each vault including 450 FSTs. This 4950 arrangement capacity was based upon the amount of the SF arising from the Paks reactors over an operating period of 10 years.

Considering the additional amount of the SF arising from the 30 year service life of the NPP reactors the overall capacity of the facility was expected to be 14 850 FSTs in 33 vaults.



1 Rail Wagon
2 Cask Handling Crane
3 Cask Preparation Area
4 Cask Transfer Trolley
5 Transfer Cask (C-30)
6 Fuel Drying And Unloading Cave
7 Roller Shutter Door
8 Lid Lifting Station
9 Load/Unload Port
10 Fuel drying tube
11 Maintenance Hatch
12 Charge Hall
13 Fuel Handling Machine Rail
14 Vault Modules
15 Storage Tubes
16 Collimators
17 Stack
18 Fuel Handling Machine

FIG. 1. Paks MVDS [1].

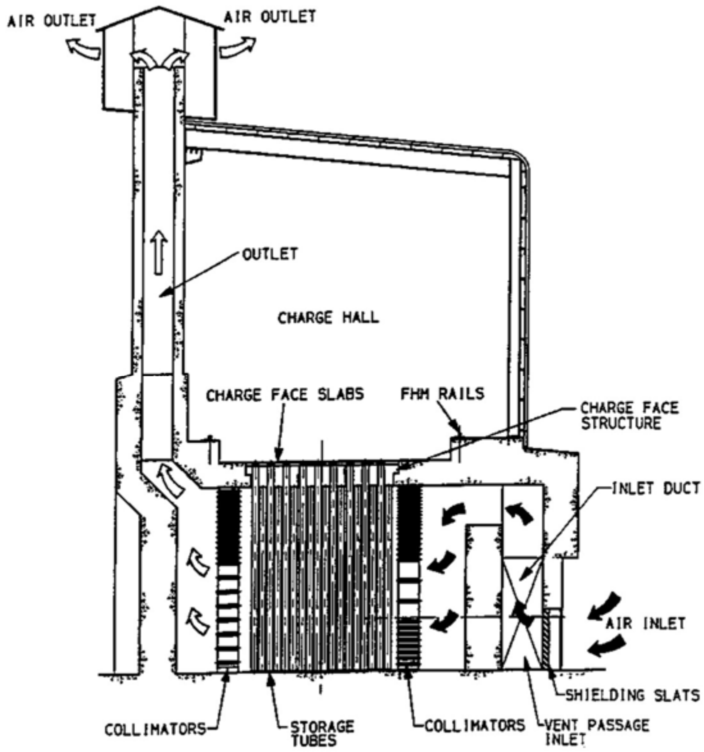


FIG. 2. Vault module schematic arrangement. [1].

As the first construction license expired further enlargement of the Paks MVDS needed to renew its license at the beginning of the 2000s. As the selection of the storage technology was carried out back in the early 1990's this time a two-step re-selection process was initiated when the aim was to make sure that the adopted technology is not just safe but economical too. In 2003 a decision was made to continue the extension of the existing storage facility using the MVDS technology.

Based on the lessons learned from the operational experience and the above-mentioned re-selection process some important modification were licensed for further enlargement phases. One of them was to increase the total storage capacity which played an important role in the life extension for the Paks NPP.

As a result of the modifications starting from the 17th vault the number of FST per vault was increased from 450 to 527. This was provided by use of the built-in reserves of the design and the development of analyses techniques which contributed to reduce the conservatism in calculations.

The constraints of this modification were the CFS structural strength and the loading machine's seismic system modification requirements. By taking into account all these conditions/criteria the overall capacity of the facility had become 16 159 (16 × 450 + 17 × 527) SFs storage capacity that has been commissioned up until now is not much more than half of what is required if the 20 years lifetime extension of the Paks NPP is also considered. Therefore, it was of paramount importance to investigate further capacity enhancement possibilities.

3. THE RATIONALE BEHIND THE CAPACITY ENHANCEMENT OF THE SFISF

3.1. Alternatives of storage capacity enhancement

To cover the required additional capacity only two possible solutions could be envisaged in terms of dry storage technology. One of the possible solutions was the deployment of dry cask storage as new technology and the other one was to further increase the capacity of the existing SFISF.

The cost analysis of the SFISF capacity enhancement made clear that reducing costs is possible by diminishing the space needed per storage tube.

According to the original design, an approx. three year minimum of cooling is applied for the SFs arriving into the SFISF.

A proposal was made to take the advantage of the fact that the SFs stored for a long period of time in the storage facility do not need the room necessary for those SFs with the initial three years cooling period prior to transporting them from the at-rector decay pool.

As more than half of the storage facility has been constructed and has been storing SFs for a long period of time already so additional modules could be designed for SFs with much longer cooling period than three years. With the rearrangement of the SFs within the SFISF the old modules could be freed up for the SFs arriving form the NPP. This rearrangement would be possible if the old SFs could be transferred to new VMs with a further increased number of FSTs.

The preliminary static analysis showed that approx. 700 FSTs could be installed into the same geometry by modifying the CFS. The proposed solution was analyzed according to the principles of interim storage relevant areas such as critical safety, decay heat rejection and radiation protection.

3.2. Critical safety

Based on preliminary model calculations with various assemblies and vault configurations it was anticipated that the justification of subcriticality would be possible.

Justifying critical safety of a denser arrangement on the other hand could benefit from the burnup credit. By taking into account the isotopic composition of the SFs significant reduction of the calculated vault multiplication factor can be achieved. Until this point critical safety calculations were carried on the basis of fresh fuel only. The denser grid arrangement proposal was explicit regarding to receive previously loaded SFs therefore fresh fuel assemblies were excluded by precondition. Thus, it was anticipated that critical safety requirements were not going to become limiting factor for the number of FSTs.

3.3. Decay heat rejection

Analysis of decay heat rejection was carried out on the basis 23 years of cooling time of SFs. According to the SFISF final safety analysis report the average heat production of a fresh SFs (approx. 3 years of cooling time) are decreased from 477 W to less than 135 W after 23 years of cooling time. Prior to the current enhancement under consideration vaults containing 527 assemblies resulted $527 \times 477 \sim 250$ kW of thermal power. Assuming for example 750 pieces of SF with 23 years of cooling time the thermal power reduces to $750 \times 135 \sim 100$ kW. As such this significant increase in the number of assemblies in one vault would result in less than half of the actual thermal load.

3.4. Radiation protection

In terms of radiation protection both operator doses and individual doses of the public had to be considered. Operator doses are made up of two effects. One of them is caused by the assemblies stored in the vaults the other is coming from the manipulation activities. The latter was considered as the decisive factor. The operational experiences regarding doses caused by manipulation activities of fresh assemblies were well below safety limits therefore no increase from the manipulations of 23 years of cooling time SFs could be expected. However, if assemblies stored in the vaults would cause an increased dose then shielding capability enhancement of the CFS could be a viable solution.

3.5. Result of the preliminary analysis

The technical feasibility analysis of the SFISF capacity enhancement was able to prove that storing all the SFs from Paks NPP of its lifetime in 33 vaults could be a realizable solution. This meant that the vaults from 25 and the facility had to be redesigned in such a way that each vault would have had to be able to receive more than 700 SFs. As opposed to this solution there was no dry cask storage technology available on the market which would have been able to match the gain projected by the SFISF enhancement considering both costs and uncertainties coming from the application of a new technology. One of the additional defining circumstances was the fact that the significant cost of soil stabilization works needed for the construction of the vault modules was previously completed to the total of 33 vault configuration.

4. DETAILED DESIGN AND SAFETY ANALYSIS

The detailed design of the facility started with the determination of the exact number of SFs needed to be stored in context with the remaining lifetime also tagging onto account the new 15 month operating cycle of Paks NPP. After an iterative decision-making process, the number of FST per vault was recorded to 703 and keeping the 33 vault configuration. That means a total storage capacity of 17 743 SFs. The concept was that SFs stored in the 1–15 vaults will be rearranged to the 24–33 vaults while the fresh SFs form the NPP will be stored in the places which thus become vacant. In order to accomplish that, 500 SFs rearrange and loading operations need to be done in the future annually.

Civil and mechanical technical plans were made with increased number of FSTs based on the previous VM design. On the bases of the technical plans detailed safety analyses were made to prove to meet the criteria defined by the facility design and legal regulations.

There are three main requirements on the design of the MVDS that had to be analysed in common with capacity enhancement for the proof of safety [2]:

— The effective neutron multiplication factor (K_{eff}) shall not exceed the value of 0.95;
— The maximum fuel clad temperature shall not exceed the value of 410°C;
— The maximum temperature of concrete structures shall not exceed the value of 100°C.

The decisive legal regulations for the operation of the facility are the followings:

— An annual risk of death to the individual of 10^{-6}/years from all radiological accidents;
— Individual operator annual dose limit (normal operation): 20 mSv;
— Offsite annual dose limit (normal operation): 10 µSv.

For the demonstration of safety critical, thermal, radiation protection analysis and probability safety assessments were required to elaborate. These safety cases are the bases of the pre-construction safety report that required for the licensing processes. The main findings of the safety cases are disclosed below.

4.1. Critical safety

The array of FSTs within the 450 FST vault of the MVDS are arranged on a triangular lattice, while in the 527 FST vault in square pitch. The square pitch was defined such that the unit cell cross-sectional area containing a single FST, was equivalent to that of the FST on the triangular pitch (325.7 mm). The array of FSTs within the 703 FST vault are arranged on a triangular lattice but with a reducing on the cell dimension to 295 mm that means a determinative parameter to critical safety.

The assessment of critical safety was based on a hypothesized flooding with potential moderators such as non-borated water aerosols in the FST and outside of it (internal and external flooding). The calculations were made with MCNPX KENO-VI. In the modelling of external flooding the interstitial water density was subsequently increased from 0.0 g/cm^3 to 1.0 g/cm^3 in the model. The maximum value obtained for K_{eff} in the external flooding occurred with an interstitial water density of 0.16 g/cm^3 giving a resultant maximum value of 0.8778±0.0003 (Fig. 3). It is interesting to note in the results of Fig. 3 that the array is in fact almost as reactive when the vault is flooded with full density water than is the case when dry.

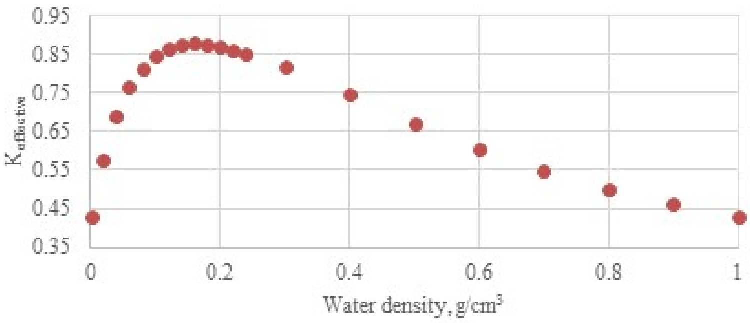


FIG. 3. K_{eff} as a function of water density in external flooding [3].

On the calculation of the internal flooding the density of the water increased from 0.08 g/cm^3 to 0.3 g/cm^3. Note that although the calculations assume internal flooding of all FSTs this is not considered to be a credible situation. The results are only to be used to assess the effect of external flooding of the vault. The value of K_{eff} was calculated to be 0.8797±0.0003 illustrating no significant change to the external model and has a substantial margin on the stated design criteria value of 0.95 (Fig. 4).

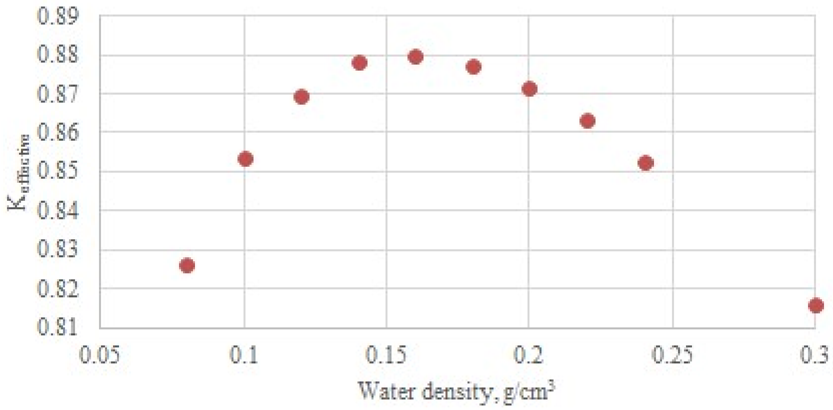


FIG. 4. K_{eff} as a function of water density on internal flooding [3].

4.2. Thermal analysis

The FSTs are cooled by a naturally-induced crossflow of atmospheric air in the MVDS. No external agent, medium or power source other than atmospheric air and gravity are required to maintain the cooling regime. The open-loop thermosyphon is achieved by means of an outlet duct extending approximately 18 m above the ceiling of the vault. The cooling air is warmed by the fuel assembly decay heat as it passes through the FST array and hence enters the outlet duct at a higher temperature than ambient. The warmed air within the outlet duct produces a buoyancy force which draws more ambient air through the vault, which in turn picks up the fuel assembly decay heat as it passes through the tube array, before exhausting through the outlet duct. The flow is self-sustaining and self-regulating in as much as the flow rate is dependent upon the total fuel assembly decay heat generation within the vault.

The thermal analysis for the new vaults was completed by using of computational fluid dynamics code (Ansys CFX 14.5). The MVDS performance for all normal and fault operating conditions has been evaluated using the maximum (1 in 100 000 year maximum) value of temperature (47.8°C). The calculation model took into consideration the case that two or more maximum irradiated fuel assemblies are in each other's neighborhood. In the worst arrangement of maximum irradiation SFs, the results showed that the FST temperature maximum does not reach 90°C even in fault situations (Fig. 5). Compared to this, in 527 FST vault the calculated maximum FST temperature was 327°C in which case the fuel clad temperature does not reach the limit of 410°C.

The maximum calculated temperature of concrete structures was 72.5°C that also a lower value than in the 527 FST vault and is below the limit of 100°C.

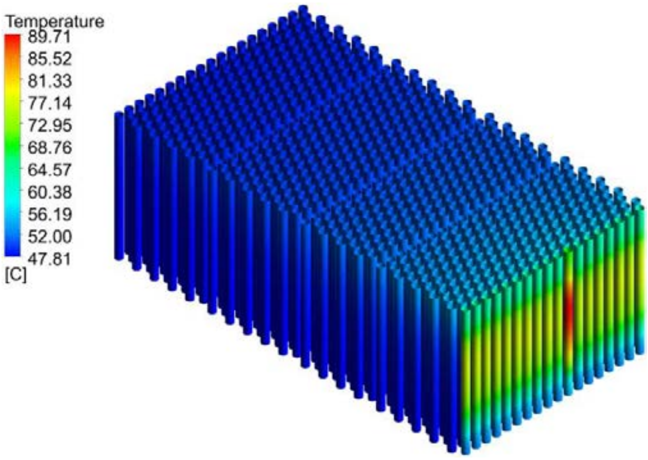


FIG. 5. FSTs temperature distribution [4].

4.3. Probability safety assessment

Probability safety assessment is an established technique to numerically quantify risk measures usually in nuclear power plants. Although SFISF is not a nuclear power plant the PSA method is well useable to rate the risk of radiological accidents. In the facility particularly SF damage accident could lead to unacceptable consequences. The final safety case report of SFISF declares that the frequency of such an event shall not exceed 10^{-7}/year. It's a stricter requirement than the Hungarian legal regulation.

The revision of the current probability safety assessment was induced not directly by the capacity enhancement of the VM but the rearranging process of the SFs. The initiating events were totally revised and one new initiating event was defined which describes unintentional loading/rearranging fresh SF to the enhanced capacity VM. In this case the above-mentioned temperature limits could be reached that would cause damages to the SF. To exclude this from the design basis events new interlocks were defined to the operability of fuel handling machine which prevent unintentional SF loading/rearranging. Taking into account the changes of fuel handling machine the results of the PSA satisfied that requirements of the safety case and regulations.

4.4. Radiation protection

The goal of the radiation protection assessments was to prove the compliance with the operator and offsite dose limits. The increase of the operator dose is caused by the enhancement of the VMs capacity, while the offside dose changes comes from the rearranging process of SFs. The acceptance criteria for the operators was the dose rate calculated at the walking surface of the CFS in the case of the 527 FST vault (approx. 10 μSv/h) [5].

The CFS is a load bearing element providing lateral support for the FST array and also a radiological shielding fabric. The CFS was an on-site concrete filled welded steel structure on the pervious VMs. The on-site concrete filling and the welding processes would not have been evolvable due to the denser FST arrangement. These problems induced the redesigning of the CFS to a completely steel framework with reduced thickness to ensure the transportability and lifting requirements. The initial radiation protection calculations showed that the neutron dose would be higher at walking surface of the CFS because of the missing concrete filling. To solve this problem the CFS was divided into two sections. The upper load bearing section was designed to a completely steel framework while the lower section became an off-side concrete filled steel structure (Fig. 6). Additional changes had to be made in the FST plug that was redesigned to a concrete filled steel element. Changes of the structures reduced the dose rates to an acceptable level at the walking surface of the CFS.

In the case of offsite doses, the detailed calculations proved that the dose rates added by the rearranging process of SFs do not have significant changes and the limits will be respected later on.

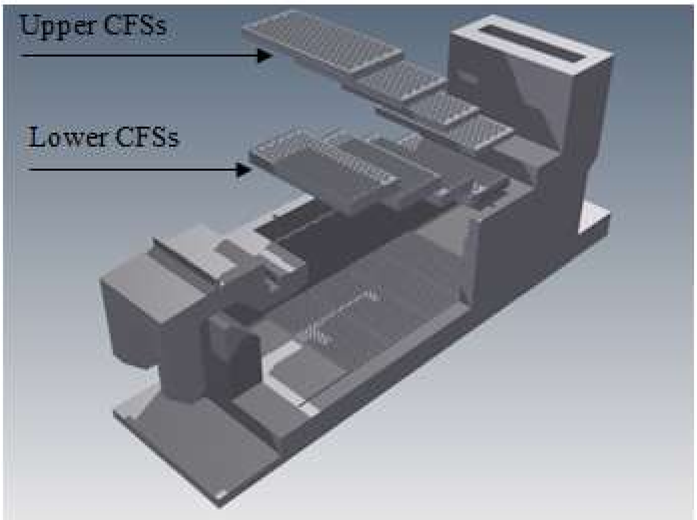


FIG. 6. Redesigned charge face structures. [6]

5. LICENSING AND CONSTRUCTION

Review of the facility-level licenses became necessary caused by planned modifications in the capacity enhancement program. According to the Hungarian legislative framework the licensing processes had to start with the renewal of the environmental license in 2015. In this process the licensee demonstrated the compliance with the limits of radioactive discharges. Next step was to prove the fulfilment of requirements set out in the Nuclear Safety Code concerning construction licensing procedure.

The authority granted construction license for the facility at beginning of 2017. Construction licensing process for the building closed successful in the end of 2017 in an individual application. In possession of the necessary licenses the project stepped further into its implementation phase.

The complete construction documentation including a 3D building information model for the next VM was completed in 2018. The Licensee intends to start the tendering processes for the construction in 2019. The construction should finish till 2024, so the commissioning and operation licensing processes could be completed in 2025.

6. SUMMARY

The 20-year lifetime extension of Paks NPP made it necessary to review the technology used for interim storage of SFs in Hungary, as the facility originally was designed for the amount of spent fuel arising from the 30 years of operation of the four Paks NPP reactors. The preliminary analysis showed that the actually applied MVDS technology with some modifications could economically provide a reliable solution to accommodate all the spent fuel by increasing the number of FST per vault but leaving the footprint of the facility according to its original size. Thus, all of the additional SFs produced by the lifetime extension could be stored in the formerly planned 33-vault configuration of SFISF.

From the safety cases of the facility it was known that the limit for the denser arrangement of FSTs comes from the relatively high decay heat production of the fresh SFs. It was recognized that the previously loaded thousands of SFs with more than 20 years of cooling time have a heat production that is much lower than the heat production of the SF newly arriving to the SFISF for storage. The idea was to rearrange the older SFs to the following modules with enhanced capacity while the fresh SFs will be loaded to the places which thus become vacant. The preliminary calculations demonstrated that a denser arrangement of FSTs is feasible in the aspects of safety and technology. The Licensee decided to execute a detailed design and analysis work to enhance the capacity of storage facility.

The most important challenge during the process was to redesign the CFS by fulfilling the structural, building technology and radiation protection requirements. A new CFS construction was created that can meet all criteria. Based on the technical plans and the revised safety cases the authorities gave permission to the construction. The development solved the storage issues of the SFs in the SFISF produced by Paks NPP after lifetime extension in the most economical way. Beyond the economic aspects it's a notable success that the capacity enhancement programme was fully designed by domestic institutions from idea to realization.

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Paper ID#10

APPROACHES TO EVALUATION OF SPENT NUCLEAR FUEL REPROCESSING PRODUCTS ACTIVITY AND VOLUME EQUIVALENCE WHICH IS RETURNED TO A SUPPLIER STATE IN THE RUSSIAN FEDERATION

A. V. KURYNDIN, A. M. KIRKIN, S. V. MAKOVSKIY
SEC NRS
Moscow, Russia
Email: kirkin@secnrs.ru

I. V. GUSAKOV-STANYUKOVICH, A. S. NOVIKOV, A. A. SHCHEKOCHIKHIN
TENEX JSC
Moscow, Russia
Email: Gusakov-Stan.I.V@tenex.ru

Abstract

According to a series of interstate agreements of the Russian Federation, spent fuel assemblies from the Russian-origin reactors are subject to return to the Russian Federation for interim technological storage and subsequent reprocessing. Meanwhile according to the legislation of the Russian Federation, products of spent nuclear fuel (SNF) reprocessing are subject to return to the Supplier's state. The principle of activity equivalent for the imported SNF and the reprocessing products returned to the Supplier's state is used in the Russian Federation to determine the volume of reprocessing products to be returned, taking into account the natural decay of radionuclides for the period of technological storage. However, there is no uniform approach to determine the activity equivalent criteria. The paper describes the main approaches implemented in the Russian Federation to determine these criteria and shows prospective ways to its definition.

1. INTRODUCTION

Today, there are tens of NPP power units with VVER-type reactors, such as the Akkuyu NPP (Turkey), the Belarusian NPP, the Ruppur NPP (Bangladesh) and others under construction or planned for construction overseas by ROSATOM [1]. In addition, a number of countries have been successfully operating power units with VVER-type reactors developed by Russian specialists [1]. For example, there are 4 power units with VVER-440 reactors in operation at the Paks NPP (Hungary), and two additional power units with VVER-1200 reactors are planned to be built [2]. The Kozloduy NPP (Bulgaria) also operates 2 power units with VVER-1000 reactors [3]. The nuclear fuel used at these NPPs is produced at the Russian enterprises and due to the complexity of the SNF reprocessing technology and its unavailability to foreign partners, in some cases, it is assumed that spent fuel assemblies (SFA) from reactors of the Russian-origin should be sent to the Russian Federation for its technological storage and reprocessing.

One of the main enterprises engaged commercially in SNF reprocessing in the Russian Federation is the RT-1 plant of the FSUE PO Mayak, that can reprocess SFA from VVER-440 and VVER-1000. One of the main products of SFA reprocessing is high level radioactive waste (HLW) vitrified in the aluminophosphate matrix [4]. Table 1 presents approximate activity values of the main radionuclides included in such HLW.

TABLE 1. RADIONUCLIDE COMPOSITION OF HLW VITRIFIED IN ALUMINOPHOSPHATE MATRIX [5].

Radionuclide	**Activity, Bq/l**
U+Pu+Np	$1.16 \cdot 10^{10}$
Minor actinides (Am+Cm)	$9.47 \cdot 10^{10}$
$^{137}Cs+^{90}Sr$	$1.01 \cdot 10^{13}$ [a]
Fission products (Sm, Sn, Ce etc.)	$4.89 \cdot 10^{10}$

[a] The activity value is given taking into account the decay products progeny (^{137m}Ba and ^{90}Y)

2. CURRENT APPROACHES TO THE RETURN OF REPROCESSING PRODUCTS TO THE SUPPLIER'S STATE

The SNF importation into the Russian Federation for the purpose of interim technological storage and/or reprocessing is carried out on the basis of the provisions of part 4 of article 48 of the Federal law of January 10, 2002 No. 7-FZ "*On environmental protection*" [6] and article 64 of the Federal law of November 21, 1995 No. 170 FZ "*On the use of atomic energy*" [7]. Under the provisions of [6, 7], the importation of SFA from foreign NPP reactors for the purpose of interim technological storage and/or reprocessing in the Russian Federation is carried out in accordance with the international agreements of the Russian Federation in the order established by the Government of the Russian Federation taking into account the basic principles of ensuring nuclear non-proliferation, environmental protection and the economic interests of the Russian Federation,, as well as the principle of priority return of radioactive waste (RW) generated after reprocessing to the state of origin of SFA.

According to requirements of the second paragraph of article 64 [7] and part 4 of article 48 [6], the Decree of the Russian Federation Government of 11.07.2003 No. 418 [8] approved the regulations on import to the Russian Federation of SFA from nuclear reactors. According to p. 17 [8], the volume of reprocessing products to be returned to the Supplier's state should be determined by the methods agreed by the parties on the basis of the activity equivalence of previously imported SFA and reprocessing products, taking into account the natural decay of radionuclides for the period of technological storage and during reprocessing. It should be noted that [8] does not contain the procedure for calculating the activity equivalent or determining the term and allows for a broad interpretation of this requirement.

Taking into account the above-mentioned variability and in accordance with the seventh paragraph of article 6 [7], the Order of Rostechnadzor of 30.12.2013 No. 655 approved the safety guide for the use of nuclear energy "*Recommendations to Safety Ensuring the Return of the Irradiated Fuel Assemblies Reprocessing Products to the Supplier's State*" RB-092-13 [9]. In accordance with the recommendations of these safety guidelines, reprocessing products should be sent to the Supplier's state in the form of solidified high level radioactive waste. Illustration of the approach to manage the return of products of SNF reprocessing to the Supplier's state implemented in the Russian Federation is shown in Fig. 1.

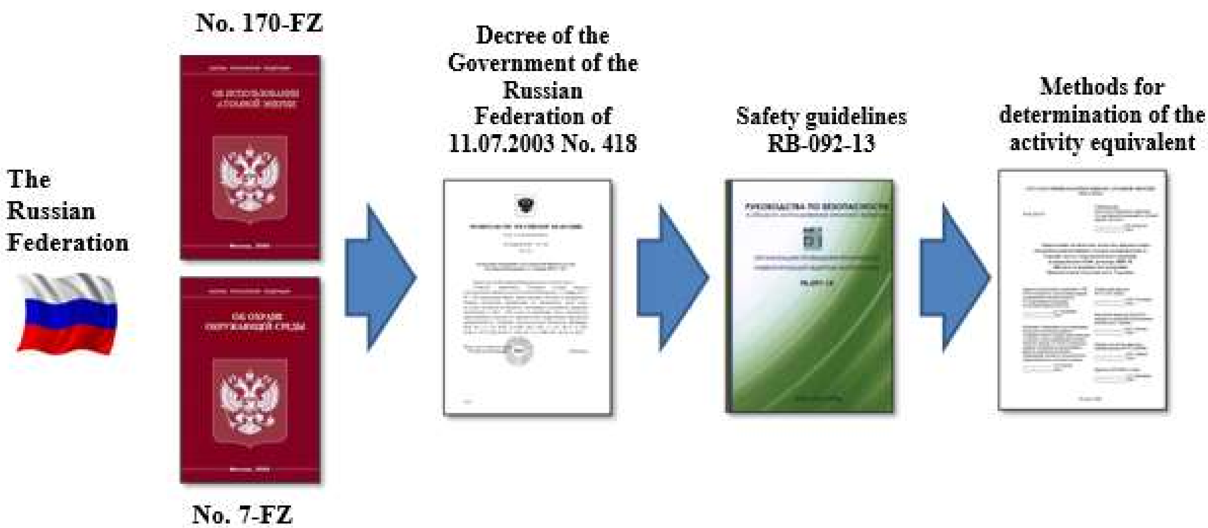

FIG. 1. Regulatory framework for the return of reprocessing products to the Supplier's state.

It is recommended to use the equality criteria of dose equivalents of SFA imported at the time of RW return and RW to be returned, taking into account the dose factors for the receipt of radionuclides with food (according to Annex 2 to NRB-99/2009 [10]) for determination of the activity equivalent and the volume of RW to be returned in accordance with the provisions of RB-092-13 [9]. The dose equivalent of a single radionuclide is determined by the formula:

$$\mathrm{E}^i(t) = A^i(t) \cdot K^i \qquad (1)$$

where $A^i(t)$– is the activity of the i-th radionuclide at time t;
K^i – is the dose coefficient of the i-th radionuclide.

It should be noted that during the development of RB-092-13 recommendations [9] a technological extraction of U, Pu and Np was taken into account as well as the fact that these radionuclides would remain in the Russian Federation as target products to be used in the nuclear fuel cycle. Therefore, the dose equivalents of these radionuclides are not taken into account in the calculations.

In accordance with the recommendations [9], it is necessary to use the following ratio to determine the SNF dose equivalent after a certain period from the moment of reactor core discharge:

$$E^{SNF}(t) = E^{FP}(t) + E^{actin}(t) + E^{AP}(t) \quad (2)$$

where E^{FP}, E^{actin}, E^{AP} – the dose equivalents of fission products, actinides and activation products.

The dose equivalent of vitrified HLW is defined as the sum of dose equivalents of all radionuclides in HLW:

$$E^{HLW}(t) = \sum_{i=1}^{k} E_i^{HLW}(t) \quad (3)$$

In most cases, vitrified HLW is understood to be HLW in the aluminophosphate matrix, the approximate radionuclide composition of which is shown in Table 1. However, the radionuclide composition of HLW may change along with the development of vitrification technologies.

The dose equivalent of HLW to be returned should be equal to the dose equivalent of SNF imported at the time of return:

$$E^{HLW}(t_{return}) = E^{SNF}(t_{return}) \quad (4)$$

This approach to determination of the activity equivalent of SFA and radioactive waste correlates with the approaches used in international practice. Thus, the Sellafield Ltd (UK) uses “toxic potential” as a criterion of the activity equivalent of radioactive substances with different radionuclide composition [11]. According to the definition given in [11], the toxic potential of radioactive material means the volume of water in which this radioactive material must be completely dissolved to such an extent that the expected dose from the consumption of water by population from this source during the year does not exceed 1 mSv/year. This approach to determining the activity equivalent of various radioactive materials aimed at comparison of the hypothetical dose effects on the population of all radionuclides contained in them when they enter the body and correlates with the approach implemented in RB-092-13 [9].

Today, there is a wide practice of using RB-092-13 [9] in determining the amount of radioactive waste returned to the Supplier’s state, including the imports of SNF from research reactors of Russian-origin.

3. NEW POSSIBLE APPROACHES TO THE RETURN OF REPROCESSING PRODUCTS TO THE SUPPLIER’S STATE

The provisions of RB-092-13 [9] should be regarded as recommendations and, according to paragraph 5 [9], the mandatory requirements of regulatory documents can be met with the use of other approaches with the validity of such approaches to ensure safety. RB-092-13 [9] recommendations were developed on the basis of an ideology that did not provide for the return to the Supplier’s state of such products as nuclear fissile materials (NFM) – regenerated uranium (RepU), plutonium (Pu), neptunium (Np). Therefore, these products are not taken into account when determining the dose equivalent of RW to be returned. It is assumed that NFM are valuable products and will be used in the Russian Federation in the form of regenerated nuclear fuel (e.g. MOX fuel).

Nevertheless, an approach is currently being worked out in the Russian Federation providing the return of products of reprocessing to the Supplier’s state in the form of fresh nuclear fuel containing regenerated NFM. In this case, there can be a significant reduction in the amount of HLW returned to the Supplier’s state as a result of taking into account the dose equivalent of regenerated NFM, included into the fresh fuel assemblies. This fact can deliver the Supplier's state from the necessity of the construction of large-scale storage and/or HLW repositories in deep geological formations on its territory which is prescribed by paragraph 1.17 of General safety requirements IAEA No. GSR Part 5 [10].

It should be noted that the export of regenerated nuclear materials does not contradict the international obligations of the Russian Federation on non-proliferation of nuclear weapons [12], as these materials are supplied in the form of fresh fuel assemblies. In addition, such option for the return of reprocessing products is implemented, for example, at the Borssele NPP (the Netherlands), which includes a single PWR-type reactor [13]. SNF generated during the operation of this reactor is sent for processing to the La Hague plant (France), and the equivalent of reprocessing products is returned to the Netherlands, including in the form of MOX fuel. At the same time, the power unit of this NPP during the first forty years of its operation was loaded with uranium oxide nuclear fuel manufactured according to classical technologies [13].

In practical terms for the Customers of the SNF reprocessing services, the above approach will lead to:

— a significant reduction of the volume of HLW to be returned, in case the Suppliers' states use regenerated NFM in their NFC;
— full compensation of the HLW activity and its staying in the Russian Federation, in case the Suppliers' states use regenerated NFM in their NFC in quantities larger than imported SNF can contain.

Today, the Russian Federation has accumulated experience in the fabrication of nuclear fuel containing regenerated nuclear materials. For example, fuel rods with REMIX fuel being operated in the VVER-1000 reactor of unit 3 of the Balakovo NPP [14]. The basis for REMIX fuel is the undivided mixture of uranium dioxide and plutonium, formed during the reprocessing of SNF from VVER reactors after its purification from other actinides and fission products. Enriched uranium is added to this mixture to create REMIX fuel. In addition, the experimental-industrial operation of the MOX-fuel production plant is being carried out at the FSUE MCC [15].

The results of a preliminary analysis of products of reprocessing to be returned to the Supplier's state considering of the return of regenerated nuclear materials in the form of fresh nuclear fuel are described below. In addition, a comparison of this option with the current approach was carried out. The results of calculation of the nuclide activity of VVER-1200 SNF with initial enrichment of 4.95 % ^{235}U and a burnup of 60 GWd/tHM were used as initial data for analysis. The calculations were performed with the use of generally recognized code SCALE [16]. The composition of vitrified HLW (see Table 1), as well as fresh REMIX and MOX fuel [17, 18], were also used as a initial data for the analysis considering the dose equivalents of the corresponding radionuclides.

It was assumed that prior to delivery to the Russian Federation, the spent fuel is stored for 10 years at foreign NPP, and the return of reprocessing products is carried out through 10 years after delivery of SFA to the Russian Federation (20 years after the reactor core discharge). The change in SNF activity during this period of time caused by radioactive decay of various nuclides (Fig. 2) is taken into account.

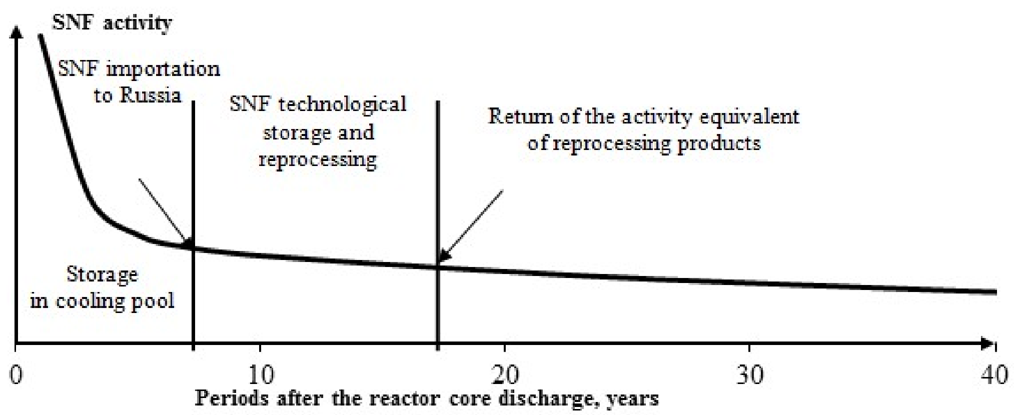

FIG. 2. Typical SNF activity dependence on the time and the main stages of SNF management.

The following ratio was used to determine the SNF dose equivalent after a certain period from the moment of reactor core discharge:

$$\mathrm{E}^{SNF}(t) = \mathrm{E}^{FP}(t) + \mathrm{E}^{actin}(t) + \mathrm{E}^{AP}(t) + \mathrm{E}^{TP}(t) \tag{5}$$

where E^{FP}, E^{actin}, E^{AP}, E^{TP}– The dose equivalents of fission products, actinides, activation products and target products (regenerated nuclear materials).

The dose equivalent of fresh fuel containing regenerated nuclear materials (REMIX and MOX fuel) is determined by the formula:

$$\mathrm{E}^{FA}(t) = \mathrm{E}^{238}(t) + \mathrm{E}^{239}(t) + \mathrm{E}^{240}(t) + \mathrm{E}^{241}(t) + \mathrm{E}^{242}(t) + \mathrm{E}^{*}(t) \tag{6}$$

where E^{238}, E^{239}, E^{240}, E^{241} and E^{242} – the dose equivalents of isotopes ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu and ^{242}Pu;
E^{*} – the dose equivalents of isotopes ^{232}U и ^{236}U (for REMIX fuel) or the dose equivalent of isotope ^{241}Am (for MOX fuel).

The dose equivalent of HLW is determined by the formula (3).
In case of implementing the activity equivalent principle for one imported SFA and fuel assembly containing reprocessed nuclear material (REMIX or MOX fuel) and vitrified HLW, the HLW mass for a single SFA (MHLW) is determined by the formula:

$$\mathrm{M}^{HLW} = \frac{E^{SFA} - E^{reg}}{E^{HLW}} \tag{7}$$

where E^{SFA}, E^{HLW} и E^{reg}г – the dose equivalents of one SFA, vitrified HLW and fresh fuel assemblies containing regenerated nuclear materials.

Table 2 shows the results of the calculations of the HLW mass, for the technology of their inclusion into the aluminophosphate matrix, and to be returned under different scenarios.
The data presented in Table 2 show that the return of the dose equivalent of imported SNF in the form of fresh fuel containing regenerated nuclear materials leads to a significant (up to 8 times) reduction in the amount of returned radioactive waste.

TABLE 2. WEIGHT OF VITRIFIED HLW TO BE RETURNED (WHEN IMPORTING ONE SFA CONTAINING 450 KG OF NUCLEAR FUEL).

Scenario of return	HLW mass to be returned		
	Return of HLW only (with the dose equivalent of NFM in SNF)	Return of HLW and one fuel assembly with REMIX fuel	Return of HLW and one fuel assembly with MOX fuel
1	2	3	4
HLW mass (kg)	2128	1848	264

The possibility of HLW fractionation with the separation of the fraction, the activity of which is formed by the isotopes of caesium and strontium (making the main contribution to the activity of HLW) is regarded as an additional direction of the development of SNF reprocessing technology and HLW management in the Russian Federation. Along with the introduction of the technology of HLW vitrification into the borosilicate matrix, due to the concentration of greater activity (up to 1000 Ki/kg [19]) in a small volume, such an option will allow:

— To form a more compact HLW matrix;
— To abandon the construction of repositories for final disposal in deep geological formations.

The generation of vitrified HLW containing only caesium and strontium has not been carried out commercially in the Russian Federation yet. Due to that fact, the exact radionuclide composition of the HLW was not determined. However, a preliminary assessment of the nuclide composition of vitrified HLW can be obtained on the assumption that all activity of the radioactive waste vitrified into the borosilicate matrix is caused only by isotopes ^{137}Cs and ^{90}Sr, as well as by their decay products. Other radionuclides are absent in the composition of the matrix. A preliminary assessment of the HLW activity is presented in Table 3.

TABLE 3. RADIONUCLIDE COMPOSITION OF THE HLW VITRIFIED INTO BOROSILICATE MATRIX.

Radionuclide	Activity, Bq/l
^{137}Cs	$4.32\ 10^{13}$ [a]
^{90}Sr	$5.76{\cdot}10^{13}$ [a]

[a] *The activity value is given taking into account the decay products progeny (^{137m}Ba and ^{90}Y)*

As part of this analysis, a preliminary calculation of the volume of reprocessing products to be returned to the Supplier's state was made, taking into account the return of the SNF activity equivalent in the form of HLW with isotopic composition specified in Table 3. Similar to the above calculations, it was assumed that the initial enrichment of SNF by ^{235}U is 4.95 %, and the burnup is 60 GWd/tHM. Table 4 shows examples of calculations of the masses of fractionated HLW vitrified into borosilicate matrices to be returned to the Supplier's state, including scenarios for the return of fresh nuclear fuel containing regenerated nuclear materials.

TABLE 4. MASS OF FRACTIONATED HLW VITRIFIED INTO BOROSILICATE MATRICES TO BE RETURNED (WHEN IMPORTING ONE SFA CONTAINING 450 KG OF NUCLEAR FUEL) [19].

Scenario of return	HLW mass to be returned		
	Return of HLW only (with the dose equivalent of NFM in SNF)	Return of HLW and one fuel assembly with REMIX fuel	Return of HLW and one fuel assembly with MOX fuel
1	2	3	4
HLW mass (kg)	667	580	87

The above results show that in case of the return of HLW only, its volume significantly reduced due to the implementation of the fractionation compared with unfractionated HLW (see Table 2). At the same time, when returning the activity equivalent in the form of fresh MOX fuel together with fractionated HLW, their amount may be less than 100 kg per one SFA imported to the Russian Federation for technological storage and subsequent reprocessing.

4. CONCLUSION

The approaches implemented in the Russian Federation to the return of reprocessing products to the Supplier's state are based on the analysis of dose coefficients of various nuclides and correspond to international approaches. Nevertheless, the Russian approach can be further improved through the involvement of the regenerated nuclear materials.

Despite the fact that the precise volumes of reprocessing products to be returned can be determined only on the basis of the SNF specific characteristics (burnup, initial enrichment, the mass of nuclear fuel, exposure time etc.), the results of preliminary analysis show that the volume of RW to be returned to the Supplier's state is significantly reduced in case of the return of the SNF activity equivalent in the form of fresh uranium-plutonium fuel (REMIX or MOX fuel). In addition, the volume of returned radioactive waste is significantly reduced when it is vitrified into the borosilicate matrix (only caesium and strontium are included).

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Paper ID#34
REDUCTION OF GEOLOGICAL DISPOSAL AREA BY INTRODUCING PARTITIONING TECHNOLOGIES UNDER CONDITIONS OF HIGH BURNUP OPERATION AND HIGH CONTENT VITRIFIED WASTE

T. OKAMURA, E. MINARI, M. NAKASE, K. TAKESHITA
Tokyo Institute of Technology
Tokyo, Japan
Email: okamura.t.ae@m.titech.ac.jp

H. ASANO
Radioactive Waste Management Funding and Research Center
Tokyo, Japan

Abstract

The thermal properties and amount of vitrified waste are major factors that determine the eventual disposal area of high level radioactive waste deep underground. The effect of high burnup operation of a light-water reactor with UO_2 fuel on the amount and thermal properties of vitrified waste under various nuclear fuel cycle conditions was discussed. In addition, the effect of Cs and Sr separation and high-content vitrified waste on reducing the waste-occupied area, which may affect the geological disposal area, under high burnup conditions was quantitatively evaluated by using the Comprehensive Analysis of Effects on Reduction of disposal Area (CAERA) index. The fuel burnup had a limited effect on the amount of vitrified waste. Furthermore, the contribution to the heat generation rate of vitrified waste for high burnup conditions of ^{137}Cs, ^{90}Sr, and their daughter nuclides, which have relatively short half-lives, increased and contribution of ^{241}Am, which has a longer half-life, decreased. Therefore, high burnup conditions reduced the waste-occupied area via Cs and Sr separation, and the maximum effect was a reduction of 74% of the waste-occupied area with a fuel burnup of 70 GWd/tHM, 4-year spent fuel (SF) cooling period, 90% Cs and Sr separation, and 30 wt % vitrified waste loading. The results suggested that fuel burnup, SF cooling period, partitioning technology, and vitrified waste loading are important for the geological disposal area, and it is necessary to consider the combination of these conditions for reducing the geological disposal area.

1. INTRODUCTION

According to Japan's 5th Strategic Energy Plan in 2018 [1], nuclear power generation is planned to provide the base load power supply in Japan. Japan's basic policy for nuclear energy use is based on the nuclear fuel cycle; thus, it is essential to develop and sophisticate the nuclear fuel cycle. However, when constructing the nuclear fuel cycle and sustaining nuclear energy use, it is necessary to dispose of a large amount of high level radioactive waste (HLW). Therefore, the reducing geological disposal area and amount of HLW are required. The thermal properties and amount of vitrified waste are two major factors that determine the emplacement of the waste packages and the eventual disposal area deep underground. These two factors, in turn, depend strongly on the radionuclide inventories contained in the HLW. The inventories mainly depend on the nuclear fuel cycle conditions, which are the fuel burnup, the spent fuel (SF) cooling period, partitioning technologies used in reprocessing, and the vitrified waste loading. Therefore, it is necessary to evaluate comprehensively and quantitatively how these conditions affect the thermal characteristics and the amount of vitrified waste generated. In particular, evaluating the effect of fuel burnup for HLW disposal is important for improving the economic efficiency and fuel utilization rate.

In this study, we quantitatively investigated the impact on the thermal properties and amount of vitrified waste of high burnup operation of a light-water reactor with UO_2 fuel under various fuel cycle conditions. In addition, we assumed that Cs and Sr separation processes and high-content vitrified waste were used for the current nuclear fuel cycle to reduce the amount of HLW and geological disposal area. The effect of Cs and Sr separation and high-content vitrified waste under high burnup conditions on geological disposal area was also discussed by using Comprehensive Analysis of Effects on Reduction of disposal Area (CAERA) index.

2. CALCULATION METHOD AND CONDITIONS

2.1. Fuel burnup conditions

The reactor operating conditions are shown in Table 1. The fuel burnup, nuclide generation in SF, decay heat generation rate, reprocessing, and partitioning were calculated with Origen 2.2-upj [2], and the cross-section libraries were based on JENDL-4.0 [3]. The fuel was UO_2 and its enrichments were assumed to be 4.5 and 6.5 wt % for burnup rates of 45 and 70 GWd/tHM, respectively. The fuel was burned assuming a burnup rate of 38 MW/tHM for 1184 and 1842 days in a 17 × 17 pressurized water reactor assembly. Fuel shuffling in the reactor operation was not considered.

TABLE 1. CALCULATION CONDITIONS FOR REACTOR OPERATION

Reactor operation	Fuel burnup, GWd/THM	45	70
	Specific power, MW/THM	38	
	Operation period, days	1184	1842
	Enrichment, wt%	4.5	6.5

2.2. Reprocessing and partitioning conditions

The reprocessing conditions were assumed to be those for a typical PUREX process after several years of SF cooling. During reprocessing, 99.6% of U, 99.5% of Pu, and 100% of volatile elements, such as H, C, I, Cl, and noble gases (He, Ne, Ar, Kr, Xe, and Rn) were removed (Table 2) [4]. The high level liquid waste (HLLW) was regarded as the residue after reprocessing. The reference case and present [5] case were studied as follows and these conditions are compared in Table 3.

(1) Reference case: the SF after discharge was reprocessed after a 4 year cooling period and no partitioning.

(2) Present study: the SF was reprocessed after a 4 and 50 year SF cooling period. It was assumed that 70% of Mo and PGM (Ru, Rh, and Pd) were separated from HLLW to satisfy the upper limit of MoO_3 and PGM loading in HLW to maintain the quality of vitrified waste and stable operation of the glass melter. In addition, the Cs and Sr separation from HLLW was assumed to be 90%. The separated elements were assumed to be immobilized and disposed of deep underground. However, the disposal area was not considered.

TABLE 2. CALCULATION CONDITIONS FOR REPROCESSING AND VITRIFICATION

Reprocessing	U, %	99.6
	Pu, %	99.5
	H, C, I, Cl, Noble gas, %	100
Vitrification	Glass weight, kg/unit	400
	Na_2O content, kg/unit	10
	Heat generation rate, kW/unit	≦ 2.3
	MoO_3 content, wt%	≦ 1.50
	PGM content, wt%	≦ 1.25

TABLE 3. COMPARISON OF SETTINGS FOR REFERENCE CASE AND PRESENT CASE

Event	Factor	Reference case	Present work
Reactor operation	Fuel burnup, GWd/THM	45	45 and 70
Reprocessing	Cooling period of SF, years	4	4 and 50
	Mo separation ratio, %	0	70
	PGM separation ratio, %	0	70
	Cs・Sr separation ratio, %	0	0 to 90
Vitrification	Waste loading, wt%	20.8	15 to 35
Geological Disposal	Waste occupied area, m^2/vitrified waste unit	41.7	13.9 to 300

2.3. Vitrification conditions

The vitrification conditions and requirements were as follows (Table 3).

(a) The weight of vitrified waste was assumed to be 400 kg.
(b) The loading of sodium oxide (Na_2O) was 10 wt % (corresponding to 40 kg) in the vitrified waste to maintain the appropriate viscosity of the melted glass in vitrification.
(c) The upper limit of the heat generation rate of the vitrified waste was assumed to be less than 2.3 kW per vitrified waste unit, consistent with the current requirement of Japan's interim storage facility [6].
(d) The upper limits of the MoO_3 and PGM loading in vitrified waste were assumed to be less than 1.5 and 1.25 wt %, respectively, to prevent yellow phase formation in the vitrified waste and deposition of PGM in the bottom nozzle of the glass melter [7–9].
(e) The vitrified waste was assumed to be stored for 50 years after vitrification to reduce the heat generation rate of vitrified waste before disposal.

2.4. Geological disposal conditions

The geological disposal site and the thermal analysis were modeled with COMSOL Multiphysics code [10]. The analytical model used horizontal emplacement of the vitrified waste in crystalline rock (hard rock), as proposed elsewhere [11]. The initial temperature, geothermal temperature gradient and thermal conductivity of the materials in the analytical model have been used elsewhere [12].

The area required for disposing of one unit of vitrified waste is called the waste-occupied area, expressed by the product of the disposal tunnel spacing (xD) and waste package pitch (y). In addition, the waste-occupied area should be set to satisfy the upper temperature limit of the bentonite buffer material. In this study, the upper temperature limit was 100°C, as used in Japan's geological disposal programs [5], and the disposal tunnel spacing, waste package pitch, and waste-occupied area were calculated at a buffer temperature of 100°C. The upper temperature limit of buffer, 100°C was conservatively assumed as temperature of illitization of bentonite which is the main component of buffer. The reason is that the performances such as water sealing property and adsorption property of nuclides required for buffer are decreased by illitization. In the reference case, the waste-occupied area was 41.7 m^2/glass unit (xD of 13.3 m × y of 3.13 m) for the horizontal emplacement configuration.

2.5. CAERA index

The CAERA index was introduced to evaluate the effect of waste-occupied area reduction under various nuclear fuel cycle conditions [13, 14]. The CAERA index was defined as:

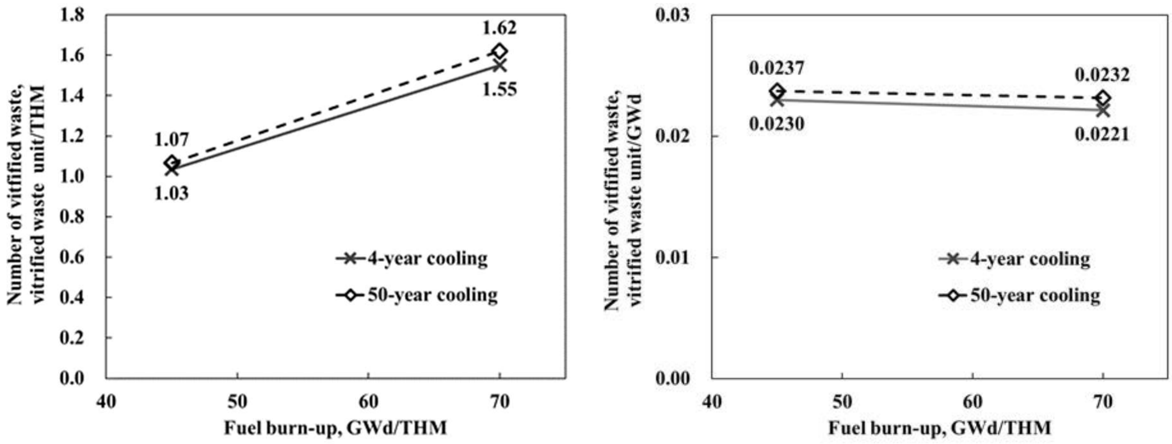
$$\mathrm{kg/m^2} = \frac{\text{Waste loading [wt \%/glass unit]} - \text{Na}_2\text{O [wt \%/glass unit]}}{\text{Waste occupied area [m}^2\text{/glass unit]}} \times \text{Weight of vitrified waste [kg]} \times \frac{1}{100} \quad (1)$$

This index has been used to evaluate the relationship between the effect of waste-occupied area reduction and partitioning technology in the nuclear fuel cycle quantitatively by comparison with a reference case [15, 16]. In this study, the CAERA index of the reference case was calculated as 1.04 kg/m^2 for a waste-occupied area of 41.7 m^2 and a vitrified waste loading of 20.8 wt %.

5. RESULTS AND DISCUSSION

3.1. Effect of high burnup operation on the amount of vitrified waste

Fig. 1 shows the relationship between fuel burnup and the amount of vitrified waste for 4 and 50-year SF cooling and a vitrified waste loading of 20.8 wt %. The amount of vitrified waste per ton of SF (glass unit/tHM) and fuel burnup (glass unit/GWd) are shown in Figs. 1 (a) and (b), respectively. The amount of vitrified waste per metric ton of heavy metal increased with fuel burnup, whereas the amount of vitrified waste per gigawatt day did not change with fuel burnup. The same trend was reported by Inagaki et al. in a similar study [17].



(a) (b)

FIG. 1. Amount of vitrified waste units per (a) metric ton of heavy metal and (b) gigawatt day (Spent fuel cooling period: 4 and 50 years; waste loading: 20.8 wt %).

3.2. Effect of high burnup operation on thermal properties of vitrified waste

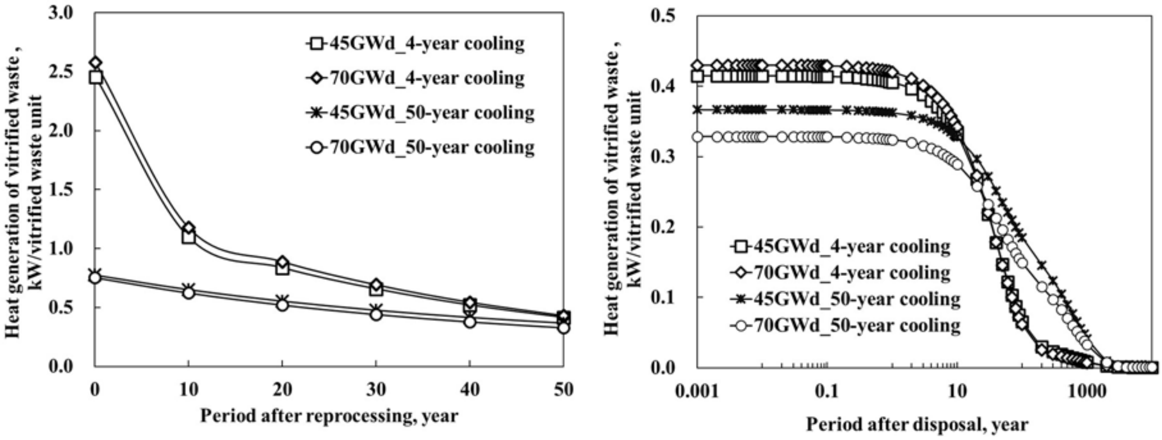
Fig. 2 (a) and (b) show the heat generation of vitrified waste as a function of time from just after vitrification to disposal and after disposal, respectively, for 4 and 50 year SF cooling and 20.8 wt % waste loading. The heat generation rates of vitrified waste from just after vitrification to disposal (Fig. 2 (a)) and just after disposal (Fig. 2 (b)) were reduced by extending the SF cooling period. In contrast, more than 10 years after disposal, the heat generation rate of vitrified waste was higher for the 50-year SF cooling period than for the 4-year SF cooling period. According to Okamura et al. [15], this is because although the contribution of Cs-137 ($t_{1/2}$ = 30.1 years), Sr-90 ($t_{1/2}$ = 29.8 years), and their daughter nuclides to the heat generation rate of vitrified waste was decreased by the prolonged SF cooling period, the contribution of Am-241 ($t_{1/2}$ = 432 years), which has a longer half-life than Cs-137 and Sr-90, was increased.

The heat generation rate was not changed by high fuel burnup for 4-year SF cooling (Fig. 2 (b)). However, for 50-year SF cooling, the heat generation rate for a fuel burnup of 70 GWd/tHM was lower than that for 45 GWd/tHM. Fig. 3 shows the contributions of Cs-137, Sr-90, Ba-137m, Y-90, Am-241, and other nuclides to heat generation immediately after disposal (50 years after vitrification) for 4-year SF cooling. The contribution of each nuclide to heat generation was different, although the total heat generation was similar. For 70 GWd/tHM, the contribution of Cs-137, Sr-90, and their daughter nuclides was larger and that of Am-241 was smaller than for 45 GWd/tHM. Fig. 4 (a) and (b) show the contribution of each nuclide to the heat generation rate as a function of the time period after disposal at fuel burnup rates of 45 and 70 GWd/tHM, respectively. The contribution of each nuclide to the heat generation rate was different for each fuel burnup rate. In addition, the contribution of Cs-137, Sr-90, and their daughter nuclides was still larger and that of Am-241 was smaller for the high burnup conditions. Therefore, the contribution of Cs-137 and Sr-90 to the heat generation rate was higher under high burnup conditions.

Fig. 5 shows the amount of Am-241 per gigawatt day in SF as a function of time after discharge from reactor for fuel burnups of 45 and 70 GWd/tHM. The formation path of Am-241 is

$$^{238}U(n,\gamma)^{239}U \rightarrow {}^{239}Np \rightarrow {}^{239}Pu(n,\gamma)^{240}Pu(n,\gamma)^{241}Pu \rightarrow {}^{241}Am \qquad (2)$$

The amount of Am-241 was affected by U-235 enrichment, based on the formation path of Am-241 and the lower weight of both nuclides at 70 GWd/tHM than at 45 GWd/tHM. In this study, the enrichment was increased to model high burnup. The U-238 content, which is the starting nuclide for Am-241 in fresh fuel, decreases with increasing fuel burnup; thus, the amount of Am-241 should also decrease. Moreover, Cs-137 and Sr-90, which are fission products, are produced more frequently under high burnup conditions. Therefore, the Cs-137 and Sr-90 loading in vitrified waste increased and the contribution of these nuclides to heat generation of vitrified waste increased with fuel burnup.



(a) Period after reprocessing until disposal *(b) Period after disposal*

FIG. 2. Time course of heat generation of vitrified waste (4-year and 50-year cooling of SF, Waste loading; 20.8 wt%).

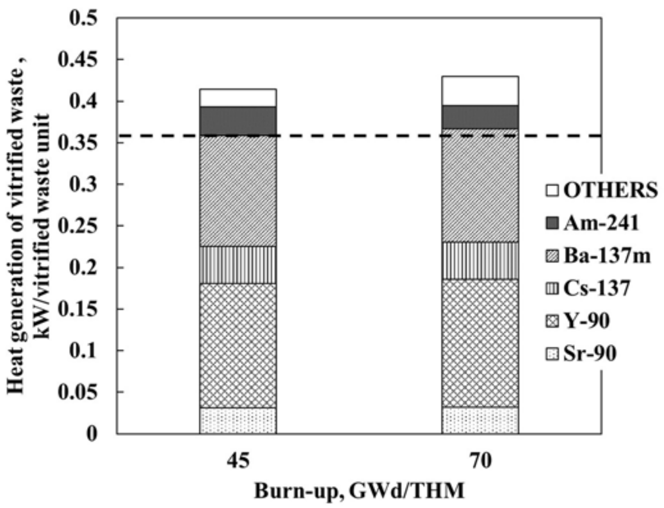
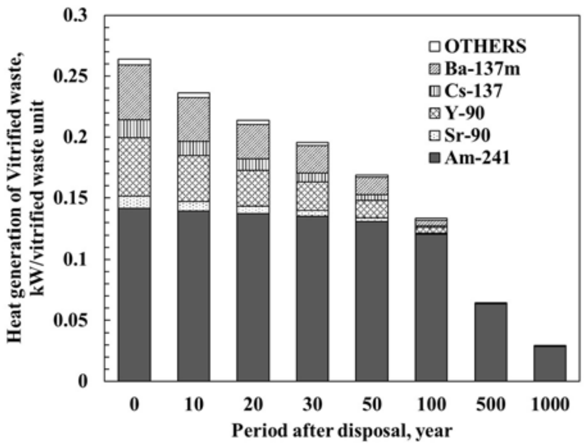
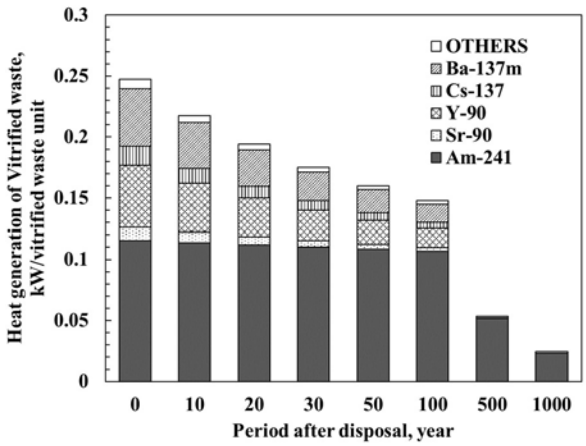


FIG. 3. Contribution of each nuclide to heat generation at disposal at fuel burnups of 45 and 70 GWd/tHM (SF cooling period: 4 years; waste loading: 20.8 wt %).



(a)



(b)

FIG. 4. Contribution of each nuclide to heat generation at fuel burnups of (a) 45 and (b) 70 GWd/tHM (SF cooling period: 50 years; waste loading: 20.8 wt %).

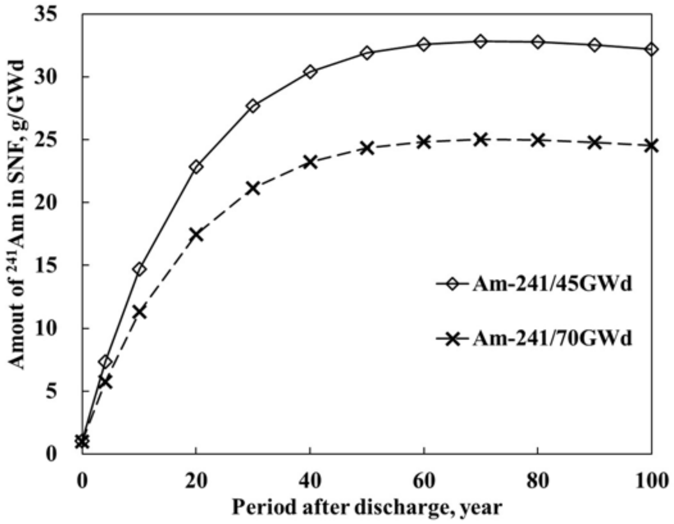


FIG. 5. Time course of amount of ^{241}Am in SF at fuel burnups of 45 and 70 GWd/tHM.

3.3. Effect of waste loading and Cs and Sr separation on waste-occupied area reduction

The effect of high burnup operation on the amount of vitrified waste and thermal properties of vitrified waste is summarized as follows.

(a) The amount of vitrified waste is not changed substantially.
(b) The contribution of Cs-137, Sr-90, and their daughter nuclides to the heat generation rate is increased and the contribution of Am-241 is decreased.
(c) The contribution of Cs-137, Sr-90, and their daughter nuclides to the heat generation rate is reduced and the contribution of Am-241 is increased by extending the SF cooling period.
(d) It has been reported that the amount of vitrified waste can be reduced by increasing the vitrified waste loading [15].

Based on these results, we investigated the effect of introducing Cs and Sr separation and high-content vitrified waste on the CAERA index for high burnup operation and shorter SF cooling period. Fig. 6 shows the relationship between vitrified waste loading and CAERA index for different fuel burnups for 4-year SF cooling and 90% Cs and Sr separation. The CAERA index was increased by Cs and Sr separation. For higher waste loading than the reference case, a substantially higher CAERA index was obtained. High burnup operation at a burnup of 70 GWd/tHM had a larger CAERA index than at 45 GWd/tHM. A maximum CAERA index of 4.00 kg/m^2 was achieved for a fuel burnup of 70 GWd/tHM and 30 wt % vitrified waste loading. This CAERA index value indicates an approximate reduction of 74% in waste-occupied area compared with the reference case (CAERA index = 1.04 kg/m2). Therefore, high burnup operation is effective for reducing the waste-occupied area by combining suitable Cs and Sr separation, SF cooling period, and vitrified waste loading conditions.

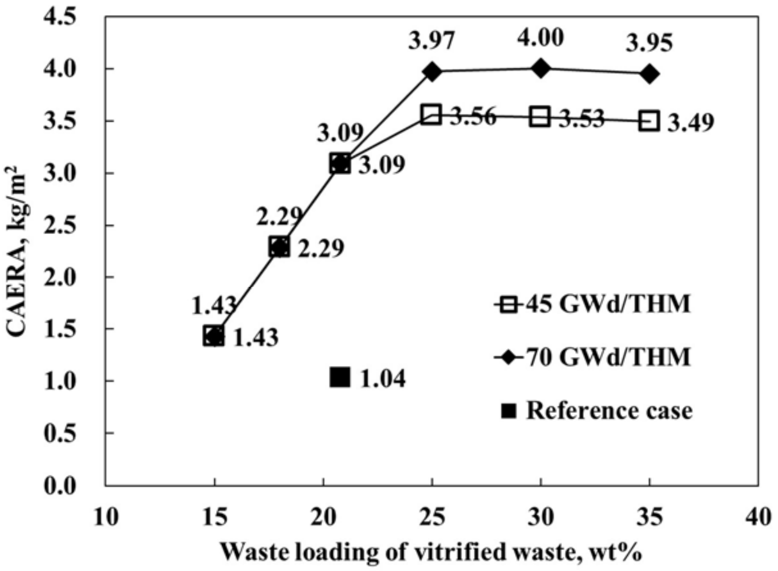


FIG. 6. Effect of Cs and Sr separation and high waste loading on CAERA index for fuel burnups of 45 and 70 GWd/tHM (SF cooling period; 4 years; Cs and Sr separation ratio: 90%).

6. CONCLUSION

In this study, we analyzed the effect of high burnup operation on the amount and thermal properties of vitrified waste. In addition, the effect of introducing Cs and Sr separation and high-content vitrified waste for reducing waste-occupied area was quantitatively evaluated by CAERA index. The conclusions can be summarized as follows.

(a) Increasing the fuel burnup did not change the vitrified waste per fuel burnup substantially, although the amount of vitrified waste per ton of SF increased.
(b) High burnup operation increased the contribution of Cs-137, Sr-90, Ba-137m, and Y-90, and decreased the contribution of Am-241 to the heat generation rate of vitrified waste.
(c) Increasing the SF cooling period decreased the contribution of Cs-137, Sr-90, and their daughter nuclides and increased the contribution of Am-241 to the heat generation rate of vitrified waste.
(d) The waste-occupied area could be reduced by introducing Cs and Sr separation and high-content vitrified waste under high burnup operation.
(e) Based on results (1)–(4), a maximum reduction in waste-occupied area of 74% could be achieved with 90% Cs and Sr separation, 30 wt % waste loading, and 4-year SF cooling.

Therefore, for high burnup operation, it is possible to reduce the amount of vitrified waste and geological disposal area by combining suitable SF cooling period, partitioning technology, and vitrified waste loading.

ACKNOWLEDGEMENTS

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STRATEGIES FOR POST-CLOSURE LONG TERM INFORMATION MANAGEMENT

J.T. HOME
Horizon Nuclear Power,
Gloucester, UK
Email: Jacob.t.home@gmail.com

Abstract

The question of disposing of radioactive waste after it has been generated is an ongoing issue for the nuclear industry. Currently one of the preferred solutions is to encase the waste in containment structures and bury it deep underground until the radioactivity has decayed to safe levels. In order to prevent future human intrusion, the repositories containing the waste much be clearly marked in a way that understandable for future society.

The paper covers the previous research efforts to develop a suitable warning system for informing future generations of the hazard posed by radioactive waste interred in a deep geological repository (DGR) or geological disposal facility (GDF) and discusses the merits a variety of approaches as well as the ethical considerations of building such a system.

1. INTRODUCTION

In 1991 a group of scientist, anthropologists, architects and science-fiction writers gathered in the New Mexico desert at the request of the United States Department of Energy (DOE) to answer a single question: how best to protect buried radioactive waste from human interference for 10 000 years?

The paper provides context for this research, discusses the methods employed by the group and their eventual findings. The paper then presents a discussion on the ethical and regulatory considerations for warning marker systems. While much of prior research has focused on US facilities, the Yucca Mountain project and the WIPP in particular, the issues discussed here are relevant to all global deep geological radioactive waste repositories.

2. THE PROBLEM

One of the best known and problematic issues facing the nuclear industry (civil and defense) is how to dispose of the radioactive waste generated through various processes. It is estimated that the worldwide inventory of waste is currently 30 million m^3 with approximately 81 000 m^3 of waste produced in OECD countries each year [1, 2]. Many of the radioisotopes generated by nuclear processes have long half-lives, the most long-lived being I-129 with a half-life of over 15 million years as seen in Table 1 [3]. In the shorter-term radioisotopes including Cs-137 account for most of the radioactivity. Actinides such as Pu-239 and Pu-240 account for a large majority of the radioactivity after the shorter half-life isotopes have decayed. For timescales over 10 000 years isotopes such as Tc-99 and Sn-126 continue to decay and produce the bulk of the radioactivity. These radioisotopes will continue to present a hazard to human health and the environment for many thousands of years after the projects and reactors that produced them have been decommissioned.

TABLE 1. HALF-LIVES OF ISOTOPES FOUND IN RADIOACTIVE WASTE

Element	T½ (Years)	Decay Mode
Caesium-137	30	γ, β-
Samarium-151	90	γ, β-
Americium-243	7370	α, γ
Plutonium-239	24 000	α, γ
Technetium-99	213 000	β-
Tin-126	230 000	γ, β-
Selenium-79	350 000	β-
Curium-247	1 560 000	α, γ
Caesium-135	2 300 000	β-
Iodine-129	15 700 000	γ, β-

3. THE SOLUTION

One of proposed solutions is to enclose the waste in vitrified form in large underground facilities known as geological deep repositories (DGR) or geological disposal facilities (GDF). These facilities will store the waste until the levels of radioactivity have decayed to acceptable levels. There are several GDF projects worldwide at various stages of development. In the US the Waste Isolation Pilot Plant in New Mexico has received HLW from the US nuclear weapons programme and the Morsleben and Schacht Asse II repositories in Germany store a mix of LLW and ILW. Further research work on deep geological disposal is being conducted in France, Australia, the UK, Belgium and Japan among others. In Finland the Onkalo facility is being constructed to receive used nuclear fuel in the form of HLW. The intention is to seal the repository in the 2120s after the final batch of waste is delivered [4].

These facilities will have to safely store their contents until the levels of radioactivity are considered acceptable. They are therefore built to last. In addition to vitrifying the waste the repositories are carefully selected based on local geology in order to lessen the probability that a future natural event such as an earthquake, erosion or ice age will damage the facility. The intention is, if left undisturbed, the GDF will withstand over 100 000 years of natural hazards to maintain its integrity. Most GDF designs share similarities of design: a series of tunnels or caverns containing the waste buried under hundreds of meters of rock [4–5]. See Fig. 1 for the layout of the Onkalo facility. After the final shipment of waste, the access tunnels would be filled in with non-porous clay. The physical site characteristics would be well understood, the site having been selected for its predictable geological characteristics. At the surface the entrance to the tunnels will be secured with physical barriers and active site controls such as alarms and security personnel.

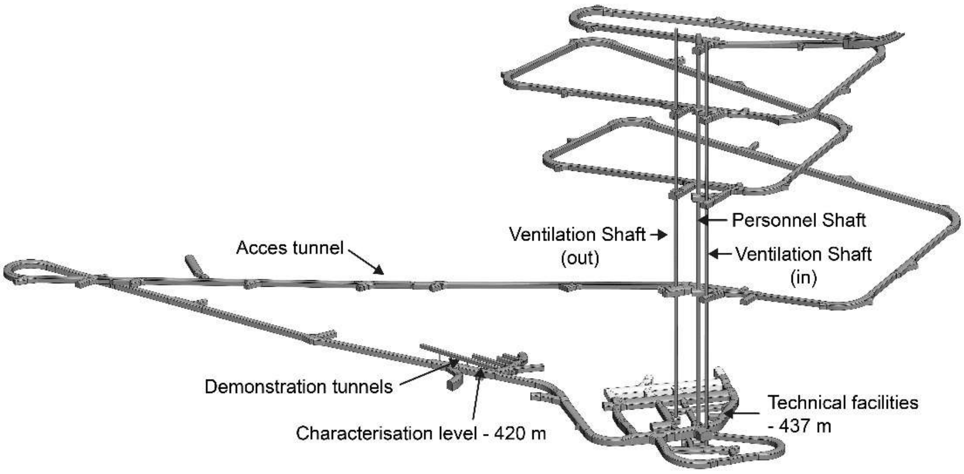

FIG. 1. Layout of the Onkalo deep geological repository [4].

4. THE PROBLEM WITH THE SOLUTION

However, less research has been done into the probability and effects of human interference with the buried waste. Either through accidental or malicious intent there is a significant probability of human interference with the GDF over its estimated 100 000 year lifetime. Due to the timescales involved it is very difficult to predict the state of the world at the end or even the mid-point of the GDF lifetime. The Future of Humanity [6] predicts that there is an up to 19% probability of human extinction by 2100 although predicting the future is notoriously difficult and the report itself cautions that "these results should be taken with a grain of salt". The report also shows probabilities for large scale deaths and disruption to human society. In the midst of these predicted societal upheavals it is very possible that institutional knowledge of the GDF and its contents will be lost. The loss does not necessarily have to happen overnight as civilization collapses. There could also be a slow roll back of funding for expensive active protection measures as the central government looks to save costs to deal with other important issues such as climate change. Information about the GDFs purpose, contents and even location could be misplaced on purpose or through decades of records mismanagement.

At this point the only barriers separating the waste from humanity will be passive physical barriers; the vitrified state of the waste, clay filled tunnels and border fences. While designed to isolate the waste from natural hazards these barriers will not withstand active human interference such as drilling. A concerted effort to mine for minerals around the GDF by a future society would have a significant probability (8.5% to 70% dependent on a number of envisioned types of future society) of impacting the integrity of the physical protection barriers [7]. It is even possible in this hypothetical future society that some information has been passed down regarding the caverns filled with mysterious treasure that the people who came before tried to hide. After all, if it isn't valuable why would past civilizations have tried to bury and hide it? There is historical precedent for this, the pharaohs of ancient Egypt initially designed their tombs to be grandiose pyramids with sealed passageways and buried secret rooms to deter curious thieves. Yet within 2000 years much of the tombs had been emptied by looters. 2000 years is only 2% of the period the GDF must keep the waste secure. The GDF must therefore be protected by a warning system which can endure and be understood by anyone who reads it in the future.

It was this problem that the US government tasked the eclectic group to solve in 1991. They were contacted as part of a study by Sandia National Laboratories, a US DOE contractor, to design a system of warning markers that could communicate the hazard of radioactive waste in a form that a future society could understand. In 1979 Congress had authorized the construction of a radioactive waste storage named the Waste Isolation Pilot Plant (WIPP) to be built near Carlsbad, New Mexico. The US Environmental Protection Agency requires that waste sites must include marker systems detailing hazards and information about the site [8] and therefore Sandia

National Laboratories was tasked to develop the design for a marker system. Within the EPA Standard (191.14) the Assurance Requirements it states that:

"*Disposal sites shall be designated by the most permanent markers, records, and other passive institutional controls practicable to indicate the dangers of the wastes and their location*."

5. REDUCING HUMAN INTRUSION

Sandia designated two panels of experts: the Markers Panel and the Futures Panel. The Futures Panel was to investigate and predict the possible paths that society might take in the next 10 000 years and the Markers Panel was to design a warning marker system that was capable of conveying information to any future society predicted by the Futures Panel. The paper concerns the efforts of the Marker Panel to develop the marker system. [9]. 10 000 years was chosen as the required time period as this was the regulatory requirement and it was considered that 100 000-year requirement was too onerous for an initial study into the markers effectiveness.

The Markers Panel was split into Team A and Team B to ensure a range of options would be generated and highlight areas where the two teams arrived at the same design or disagreed on the effectiveness of other designs. These comparisons would form the basis of further investigative work. The remit given to both teams was as follows [9]:

— The time frame for the Panel to consider must be 10 000 years because of the requirement that performance assessments cover a period of 10 000 years after closure of the disposal facility;
— The markers must be developed with a goal of being able to convey information to any future society (considering the broad spectrum of possible future societies developed by the Futures Panel [8]);
— To communicate the dangers associated with the waste buried at the WIPP.

The two teams presented their findings to Sandia National in the 1992 report "Expert judgement on inadvertent human intrusion into waste isolation pilot plant" [9]. Both teams assumed that there is potential for much change over the next 100 000 years and it is possible that knowledge of the GDF and what it contains may be lost. The languages spoken are also likely to change significantly so the messages cannot be written only in English, or any other language currently in use. In order to convey the content of the warning the message must be designed to communicate at a level beyond written alphabetical language.

If the message is to remain during the lifetime of the GDF it must be comprised of erosion resistant materials or located underground to preserve it. The material should not be considered a valuable or useful resource in case it is looted or repurposed for building material. The message must be capable of conveying 3 parts:

— That there is a message at all;
— That hazardous substances are located in this area;
— Information about the hazard.

Therefore, there is balance between the simplicity of the message which would allow it to be understood more easily and the complexity of the information contained in it which is required to describe the nuclear waste. A more complex message may have to rely on scientific prerequisites which may not exist when the message is read. Conversely, a simple message may rely too much on contextual cues or be unable to convey all the necessary information. While language is useful for transmitting specific information, it is heavily dependent on specific cultural context and knowledge so an ideal system would make use of both language and signs. The context for the reader of the message is unlikely to be the same as the designer.

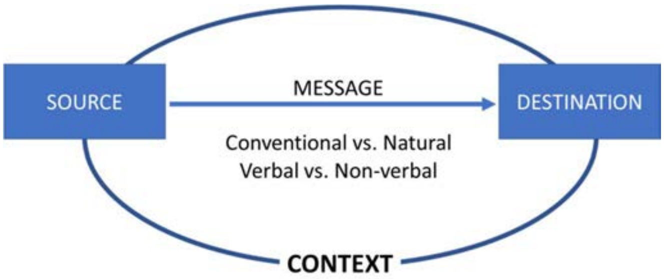

FIG. 2. Types of messages in relation to context [10].

As shown in Fig. 2 the context in which the message is received will affect its interpretation. As context changes rapidly, even daily the two teams examined methods of communication that were less reliant on context than language. Both teams recommended the use of symbols or signs as an effective method of communication. There are several ways to classify signs dependent on their reliance on context and the complexity of information they can convey; symbolic, indexical or iconic. Fig. 3 shows examples of the three classes of signs and how the signifier message is related to the signified information. However, some signs are also heavily reliant on cultural context. Generally, a sign with significant cultural attachments (a symbol) can convey a lot more information than an iconic sign which does not rely on contextual cues. Iconic signs are therefore more likely to remain understandable for longer but are limited in the complexity of information they can convey.

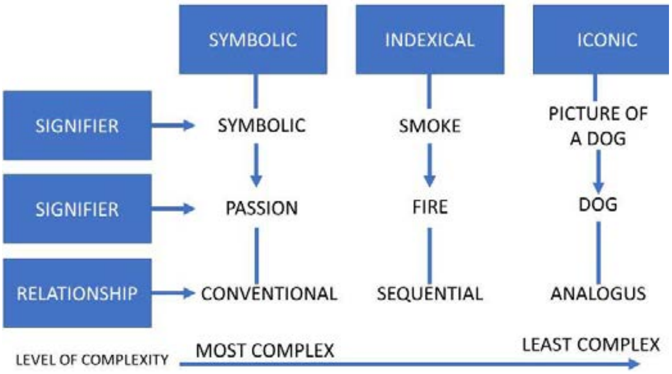

FIG. 3. Classes of signs and relationship between the signifier message and the signified information [10].

Iconic messages do not rely on contextual cues but can be limited in the information they can carry. They have a physical resemblance to the signified meaning of the message. The iconic message shown in Fig. 4 is a simple pictograph created from a previous study by MF Kaplan to design a warning message for the Hanford Site in Washington which stores transuranic waste. The diagram was meant to show the location of buried radioactive waste.

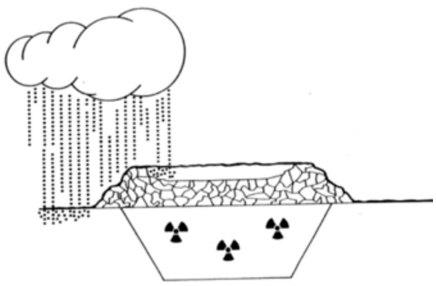

FIG. 4. Example of an iconic message showing the location of buried material [11].

An indexical message can show the connection between the physical form of the message and its meaning e.g. a picture of radioactive waste and its effect on the human body. Fig. 5 shows one of the Sandia teams proposed series that would be read vertically downwards showing the effect of radiation on the human body.

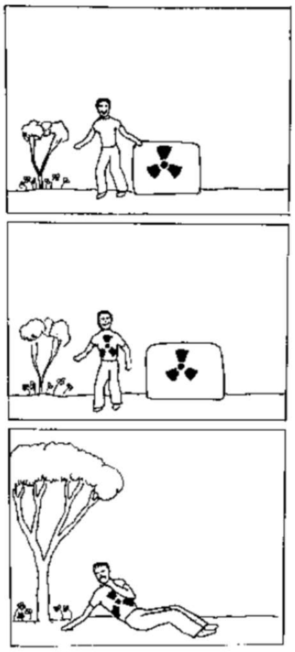

FIG. 5. Example of an indexical message showing the effect of radiation on the human body [9].

Symbolic messages are capable of carrying a lot of information but do not resemble the signifier that is being represented. They are learnt culturally and rely heavily on context. Symbols are widespread in our culture from the Golden Arches to the hammer and sickle to the skull and crossbones. Each of these symbols convey an array of meanings depending on the reader and where it is seen. However, these are relatively recent meanings in comparison to the expected lifetime of the GDF. The skull and crossbones in particular has had a variety of meanings from its origins in medieval paintings to piracy to denoting poisonous substances. It is very likely that well known symbols like these will continue to evolve over the centuries. Therefore, they cannot be relied upon to accurately convey information to future society.

FIG. 6. The skull and crossbones/Jolly Roger. An example of a symbol with changing meaning.

One of the other proposals by the teams was the use of human faces and expressions to convey information. While many symbols and signs will lose their effectiveness over time the human expressions of pain, fear and disgust are likely to remain effective ways to communicate danger and hazards. There was disagreement among some team members on the level of emotional weight the message should carry. They believed that even if future humans read and understood the information in message they may choose to ignore it unless there was an emotional component to the message as well. They recommended that any written message include a stark warning to the reader:

> *"Sending this message was important to us. We considered ourselves to be a powerful culture. This place is not a place of honour…no highly esteemed deed is commemorated here… nothing valued is here. What is here is dangerous and repulsive to us. This message is a warning about danger"* [9].

The final recommendations from the two teams were to create large earthen berms to designate the area around the GDF which would contain monoliths inscribed with the above message in the 6 languages of the UN as well as the local Navajo language. The marker system would have several different components to ensure redundancy and 'defense-in-depth'. These included buried message discs or capsules made from durable but worthless materials such as clay, stone markers in the sealed tunnels, a world map of other disposal sites and multiple buried information chambers. These chambers would include information about the marker system and GDF in several levels of complexity. The two teams differed on the opinion of whether to direct visitors focus to the information with team A advocating no sense of center ("nothing is here") and team B recommending that visitors be directed to the center to provide information about the site [9].

7. HUMAN INTERFERENCE TASK FORCE

The Markers Panel was not the first attempt to design a warning marker system for a nuclear waste repository. In 1981 the US government recruited a variety of experts to form the Human Interference Task Force (HITF) for the purpose of investigating how to reduce the likelihood of humans intruding on the Yucca Mountain nuclear waste repository. The HITF generated several proposals for how a message might be communicated:

— Representative pictograms on stone markers or monuments around the repository;
— A series of clearly artificial earth berms surrounding a central vault which contains relevant information;
— Hostile architecture" around the site to deter intruders;
— A small, sheltered group of scientists who maintain knowledge of the repository regardless of events in the wider world; an "atomic priesthood";
— Genetically engineered animals and plants such as cats or cacti that alter colour in the presence of high radiation levels to alert local people to the threat of waste leaking;
— Security in obscurity: Make the surface of the site as plain as possible to avoid future generations investigating the site [12].

8. ETHICAL CONSIDERATIONS

While creating a permanent marker system at the WIPP was required by regulations those standards do not apply to other countries building GDFs. Some countries will not have the same requirements to warn future generations and it brings another factor into consideration. The costs involved with the design, construction and maintenance of any maker system robust enough to endure for 10 000 years are likely to be substantial. A recent paper by Van Luik et al [13] noted that the cost-benefit calculation for how many lives a marker system would save would be a very difficult enterprise. Even with the input of the Futures Panel the make up of any future society is problematic to estimate. For every prediction of societal collapse there is another where technological advance continues and radioactive waste is no longer a significant hazard or can be repurposed. In this instance the marker system would be an expensive and superfluous landmark. Hora et al [7] predicted that the worst-case intrusion scenario was from resource miners with 1800-level drilling technology and no knowledge of radioactive hazards. Even in this scenario the likelihood of drilling equipment damaging the waste drums and the estimated radiological release was low [7]. There is a reasonable argument to be made that the funding required to construct the marker system would be better used investing in local infrastructure around the GDF such as roads, hospitals and schools. This will not only improve the local stakeholders opinions on hosting a GDF in their area but may also contribute a stabilizing effect to the local society to help it better resist any events that would cause knowledge of the GDF to be lost. Local communities would understandably be unhappy if it appears that future, unnamed people are being afforded more protection then them. Therefore, if there is no regulatory requirement for the warning marker system any GDF project must seriously consider the costs and benefits of having such as system.

9. CONCLUSION

Currently the WIPP site still has active control measures and a final decision has not been reached regarding the form of the marker system. Since 1991 assessment no further large-scale studies have been performed to design a system capable of communicating across 10 000 years. As nuclear reactors worldwide continue to operate and be decommissioned the quantity of nuclear waste will continue to grow. Assuming there is not technological solution found the GDF remains the best option for managing this waste. In the US there is a regulatory requirement for a marker system to warn future generations up to 10 000 years in the future of the hazard. However, it is important to consider the contemporary cost in relation to the potential future benefits when constructing the warning marker system. What is required now is further development of the design and analysis of the most cost-effective marker system taking into account future generations.

ACKNOWLEDGEMENTS

The author would like to thank Horizon Nuclear Power for providing the opportunity to present at this conference and Daniel Galson for providing expert information on reducing the likelihood of human interference to the repository.

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4. CLOSING SESSION

4.1. CHAIRWOMAN'S CLOSING REMARKS

Closing speech as provided, verbatim.

Susan Y. Pickering
Director Emeritus, Sandia National Laboratories, USA

Good morning!

I must begin by thanking my Programme Committee and the IAEA Secretariat, especially Amparo González-Espartero and Laura McManniman. This was a great team! Thank you for making this spent fuel management conference an overwhelming success!

We have had 258 participants, representing 45 Member States. We had 92 oral presentations, 31 posters and 14 E-Posters. Many opportunities to learn from the past. We learned at multiple scales – from far reaching national strategies to detailed fuel rod failure mechanisms. We gave our future leaders an opportunity to learn. Thirty-five young generation members gave an oral or poster presentation.

Compared to the last spent fuel conference, held 4 years ago, this conference was different in two significant ways.

1) The presentations had a much stronger sense of urgency about the need to successfully implement final disposal, whether for open or closed fuel cycles. Finland, you are leading the way!

2) The second significant difference from the last conference was you, the diversity of the participants. Your interest topics were broader and included storage, transportation, disposal, recycling, safeguards and economics. You are researchers, operators, regulators, government officials and academics. You are also very strong. We were surprised when you stayed until 18:30 or 19:00 every evening, asking questions of our presenters!

Several recurring themes emerged over the week.

The value of sharing data and operational experience was evident in every session. Your presentations clearly indicated collaborations lead to better solutions, whether they be happening now — like the IAEA Underground Research Laboratories Network for Geological Disposal, the high burnup demonstration cask collaboration, and the efforts of the European Commission on advanced fuel cycles including partitioning and transmutation — or collaborative efforts still in the conceptual phase — such as a multinational repository. The IAEA provides many opportunities for newcomers to learn from mature programmes.

The value of data and modelling to improve decision making was another theme. Research in ageing management is improving monitoring and inspection techniques. Thermal analysis of the high burnup demo cask may lead to re-evaluating safety margins and avoiding excessive costs.

This week we saw how important understanding the integrated nature of the fuel cycle is. For example, former fuel designs may not allow failures to be repaired, parameters such as fuel type and burnup affect geological repository design such as waste package pitch and tunnel-to-tunnel distance. We explored many questions such as, how will accident tolerant fuel affect the back end of the fuel cycle? How can safeguards be incorporated into the design of the facility?

This week we learned how the long lifespan of nuclear facilities drives decisions and actions such as R&D, operations, knowledge management and the need to develop our next generation. In the nuclear fuel cycle, a lot of time can pass between a problem and its solution.

Many presentations spoke to the benefits of following standards and guides. One even identified the need for a best practice guide to estimating costs for geological repositories.

Costs and a lack of sustainable funding were a concern for many participants. Costs and risks were explored. The concept of portfolio affordability was presented and asked: when would it make sense to switch from direct disposal to recycling? The benefits of commercial funding were discussed with respect to the Yucca Mountain Project and THORP at Sellafield. The pressures to manage costs were also mentioned.

Several innovative approaches were presented to close the fuel cycle with thermal reactors and multirecycling. We even learned of the valuable benefits of using by-products from partitioning for treating eye cancer! Several countries are exploring using partitioning to reduce the heat load of high level waste and reduce repository volume.

The critical role local communities play was a strong theme. Lessons were shared about the different approaches applied by WIPP and Yucca Mountain Project, and the very different results the approaches produced. Successes from Canada, Finland and Sweden were shared and could be models for stakeholder engagement.

Public understanding, and hopefully acceptance, is our #1 driver. Politics is a close second. If politics makes the decision, we, the people in this room, can get it done!

Thank you for sharing your lessons.

I hope the conference has given you useful learnings for your future journeys.

4.2. INTERNATIONAL ATOMIC ENERGY AGENCY DEPUTY DIRECTOR GENERAL NUCLEAR SAFETY AND NUCLEAR SECURITY’S CLOSING REMARKS

Closing speech as provided, verbatim.

Juan Carlos Lentijo
IAEA Deputy Director General, Department of Nuclear Safety and Security

Dear delegates, dear Madam Chair,

Thank you for this opportunity to close this conference. My busy schedule prevented from attending as many sessions as I would have liked, but I followed several with great interest. And I was impressed.

The conference has benefited from high attendance. Combined with interesting presentations, vibrant discussions and fruitful exchanges of experiences, this made for a successful conference with useful outcomes.

At the Agency, we have noted that there is strong interest in this conference, and that holding it regularly brings several benefits. This conference is the fourth of its kind, with the most recent held in 2015. The conference series allows all involved to regularly review progress made in the safe, secure and sustainable management of spent fuel. Holding the conferences regularly also enable the spent fuel community to keep the momentum and underline the message that the safe, secure and sustainable management of spent fuel is a must – not only because it is the right thing to do, but also because it is the key to the future of nuclear energy. In addition, regular conferences on spent fuel management help the Agency develop and adjust its activities related to this topic.

The Agency’s activities include the development of safety standards. The standards serve as a global reference for protecting people and the environment and contribute to a harmonized high level of safety worldwide. We develop the standards together with Member States in a thorough and consultative process that results in consensus. We have standards on all relevant topics, including, of course, spent fuel management.

We also help countries apply the standards, for example by offering peer reviews and advisory services. These include the Integrated Regulatory Review Services, or IRRS, the Operational Safety Review Team service, known as OSART, and of course the Integrated Review Service for Radioactive Waste and Spent Fuel Management, Decommissioning and Remediation, or ARTEMIS.

In addition, the Agency’s Coordinated Research Projects and technical reports that gather best practices, operational experiences and lessons learned help countries improve safety by applying the safety standards.

The Agency also promotes international legal instruments related to nuclear safety and security, and we help countries fulfil the obligations of the instruments. These include, notably, the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management. By living up to their obligations under the Joint Convention, its parties contribute to a higher level of safety worldwide in the management of spent fuel and radioactive waste. The Review Meetings, held every three years, play an important role in maintaining the momentum.

Dear delegates,

This conference had the subtitle Learning from the Past, Enabling the Future. And I think we have. By learning from each other at occasions such as this conference, we are together enabling a better future for spent fuel management – that is, a future in which management of spent fuel is safe, secure and sustainable.

To conclude, I would like to thank all participants for active participation in this conference and the scientific secretaries for the effective organization of the event. And I in particular thank Susan Pickering for her leadership in chairing the conference and ushering its success.

I hope to see you again in four years for the next conference on the management of spent fuel.

Thank you.

ANNEX I: CONFERENCE STATISTICAL DATA

Prepared by Conference services IAEA

International Conference on the Management of Spent Fuel from Nuclear Power Reactors: Learning from the Past, Enabling the Future

Vienna, Austria

24 – 28 June 2019

Organized by:	NEFW-Nuclear Fuel Cycle and Materials Section NSRW- Waste and Environmental Safety Section	
Location:	HQ, C Building	
Cooperating Organizations:	European Commission, OECD/NEA, WNA	
Total no. of participants and observers:	268 (24% female, 76% male)	
…thereof, officially designated participants:	232 (MS) and 26 (Orgs)	
… thereof, observers:	10	
No. of countries:	45 MS	
No. of organizations represented:	7	
No. of statements / presentations and others:	Sessions	19 (incl Opening and Closing)
	Presentations/ Speakers	101
	E-Poster presentations	14
	Posters (in 7 Sessions)	48
Scientific Secretaries:	Amparo González-Espartero, NEFW Gerard Bruno, NSRW	
Scientific and Administrative support:	Laura McManniman, NEFW Sakura Gyay de Goyaz, NEFW	
Conference Coordination:	Martina Neuhold, MTCD Sanjai Padmanabhan, MTCD	
TC Coordination:	Jing Zhang, TCEU	
Conference website:	https://www.iaea.org/events/management-of-spent-fuel-conference-2019	

- **No. of participants by Member State: 232**

Argentina	1	Hungary	4	South Africa	1
Armenia	2	India	4	Slovenia	2
Azerbaijan	1	Indonesia	4	Slovakia	3
Belgium	4	Iran, Islamic Republic of	1	Spain	8
Bangladesh	1	Italy	1	Serbia	1
Brazil	2	Japan	17	Sudan	1
Bulgaria	1	Kenya	2	Sweden	3
Belarus	3	Lithuania	1	Switzerland	5
Canada	5	Malaysia	2	Syrian Arab Republic	1
China	15	Pakistan	1	Tajikistan	1
Czech Republic	2	Philippines	1	Turkey	3
Egypt	7	Poland	2	United Arab Emirates	3
Finland	7	Korea, Republic of	7	United Kingdom	13
France	28	Romania	4	Ukraine	3
Germany	16	Russian Federation	16	United States of America	22

- **No. of participants by Organization: 26**

EC (European Union)	5	OECD (Organization for Economic Co-Operation and Development)	1
EPRI (Electric Power Research Institute)	3	WNA (World Nuclear Association)	4
IAEA (International Atomic Energy Agency)	10	WNTI (World Nuclear Transport Institute)	2
ISO (International Organization for Standardization)	1		

ANNEX II: SUPPLEMENTARY FILES

The supplementary files for this publication can be found on the publication's individual web page at www.iaea.org/publications

TITLES AND AUTHORS OF PRESENTED PAPERS AND POSTERS INCLUDED IN THE SUPPLEMENTARY FILES AS THEY APPEAR IN THE CONFERENCE PROGRAMME

TRACK 1: NATIONAL STRATEGIES FOR SPENT FUEL MANAGEMENT

ID#194 European Commission's Joint Research Centre research on the safety of spent fuel and high level radioactive waste management
M. Martín Ramos, V.V. Rondinella, T. Wiss, D. Papaioannou, R. Nasyrow

ID#23 Kenya's policy and strategy on radioactive waste and spent fuel management
H. Mpakany, V. Mutava

ID#20 Legislative framework for spent fuel management in Indonesia
S. Prihastuti, D. Sinaga, D. Taufiq

ID#4 The need for managing spent nuclear fuel in Brazil
A. Soares, P.F. Frutuoso E Melo

ID#80 Spent fuel management strategy development in Belarus
A.V. Kuzmin, S.N. Sikorin, A.P. Malykhin, V.T. Kazazjan, M.L. Zhemzhurov, T.K. Hryharovich

ID#143 Capacity extension of ISFSF Jaslovske Bohunice
J. Václav

ID#101 Spanish national strategy for SF & HLW management. Considerations for a Centralized Storage Facility (CSF)
F. Lentijo, P. Gallego

ID#204 French nuclear fuel cycle
E. Touron, C. Evans, J. Van Der Werf

ID#74 Spent fuel management - India
J.S. Yadav, K. Agarwal

ID#171 The strategy of closed nuclear fuel cycle based on fast reactor and its back end R&D activities
Y. Guoan, Z. Weifang, H. Hui, Z. Hua

ID#180 Lessons learned from the U.S. national strategy – a personal perspective
P. B. Lyons

ID#58 Japan's nuclear fuel cycle policy
K. Yoshimura

ID#160 THORP – commercial reprocessing at Sellafield
P. Hallington

ID#25 SNF management in Russia: Status and future development
A.V. Khaperskaya, O.V. Kryukov, K.V. Ivanov

ID#189 Spent fuel and radioactive waste management in Armenia
V. Keshishyan, A. Petrosyan

ID#183 Strategy of spent nuclear fuel management to NPP in the Republic of Belarus
V. Paliukhovich

ID#192 The management of spent fuel from nuclear power reactors learning from the past, enabling the future: A strategy/context for the Republic of South Africa
S.P Bvumbi

TRACK 2: SPENT FUEL AND HIGH LEVEL WASTE STORAGE AND SUBSEQUENT TRANSPORTABILITY

ID#22 The history and future plans for the wet and dry long term spent fuel storage experiments at Canadian Nuclear Laboratories
A. Barry

ID#173 The PWR spent fuel dry storage project experience feedback in China
S. Bin, Y. Chen, R. Liao,C. Cheng

ID#60 Implementing ageing management programme in interim wet storage
M. Nyström

ID#88 Update to the long term storage of advanced gas cooled reactor (AGR) fuel
A.D. Ledger

ID#104 Industrywide global efforts toward long term monitoring of neutron absorber materials in spent fuel pools
H. Akkurt, E. Wong

ID#115 Research towards prolonged interim storage from the regulatory body perspective
M. Schwerdtfeger, C. Drobniewski, C. Borkel, C. Gastl, C. Bunzmann

ID#27 Development of helium leak detection methods for canisters (part 1). Evaluation of minute gas leaks from canisters by small-scale models
H. Takeda, S. Okazaki, M. Goto

ID#103 Aging management of dry storage systems Centralized Interim Storage facilities
P. Narayanan

ID#162 Robotically-deployed NDE inspection development for dry storage systems for used nuclear fuel
J.B. Renshaw, J. Beard, J.J. Stadler, S.M. Chu, N. Muthu, M. Orihuela

ID#182 Lessons learned from Fukushima Daiichi nuclear accident for spent fuel storage
K. Shuji, T. Ritsuro, F. Takatoshi

ID#134 Borosilicate glass HLW stability during long term interim storage
C. Roussel, S. Peuget, E. Reigner, F. Frizon, L. Gagner

ID#3 Scientific basis of thermal safety analysis of dry storage of spent nuclear fuel on Zaporizhska NPP
S. Alyokhina

ID#7 Transportation cask and concrete module design for managing nuclear spent fuel produced in Bushehr Nuclear Power Plant
E. Bayat, A.M. Taherian, A.H Feghhi, F. Golfam

ID#147 Nuclear spent fuel storage: Concepts and safety issues
F. Ledroit, A.C. Jouve, I. Le Bars

ID#164 Thermal 324odelling Round Robin of the high burnup demonstration cask
A. Csontos, K. Waldrop, S. Durbin, B. Hanson, J. Broussard, G. Lenci

ID#83 A proposed regulatory perspective on the security requirements for spent nuclear fuel
Z. F. Hassan Akl

ID#190 Overview of regulatory framework on dry cask storage in Japan
T. Nishizaki, T. Morimitsu, K. Aoki, H. Yamagata

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D. Papaioannou, R. Nasyrow, R. Gretter, L. Fongaro, J. Somers, V. Rondinella, E. Vlassopoulos, S. Caruso

ID#86 PWR spent nuclear fuel integrity evaluation results in Korea
D. Kook, Y. Koo

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ID#114 Authority experience during design approval procedure for packages loaded with special encapsulations for damaged spent nuclear fuel
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A. Peršič, T. Nemec, T. Bajcar

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O. Dudka, I. Bilodid, Y. Kostyushko

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J.B. Renshaw, S.M. Chu, A. Csontos

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ID#110 An integrated approach to aging management of spent fuel dry storage systems in the United States
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ID#146 French "Cycle Impact" approach. Integrated approaches to the back end of the fuel cycle
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