Volume XLIV, Number 4

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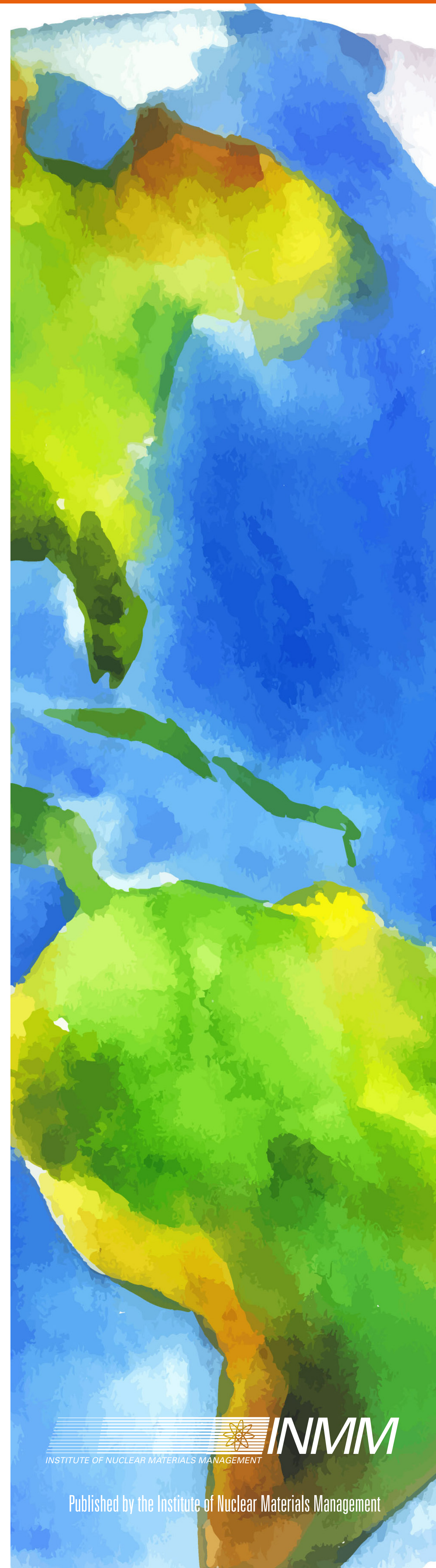


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Communicator
The Institute of Nuclear Materials Management

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INSTITUTE OF NUCLEAR MATERIALS MANAGEMENT

57th Institute of Nuclear Materials Management Annual Meeting

July 24 – 28, 2016

Atlanta Marriott Marquis
Atlanta, Georgia USA

www.inmm.org

JNMM (ISSN 0893-6188) is published four times a year by the Institute of Nuclear Materials Management Inc. The Institute of Nuclear Materials Management (INMM) is an international professional society dedicated to development and promulgation of practices for the safe, secure and effective stewardship of nuclear materials through the advancement of scientific knowledge, technical skills, policy dialogue, and enhancement of professional capabilities.

DIGITAL SUBSCRIPTION RATES: Annual (United States, Canada, and Mexico) $200 for individuals. Institutional subscriptions are $500 per year. Single copy of the proceedings of the Annual Meeting (United States and other countries) $200. Send subscription requests to *JNMM*, One Parkview Plaza, Suite 800, Oakbrook Terrace, IL 60181 USA. Make checks payable to INMM.

DISTRIBUTION and delivery inquiries should be directed to *JNMM*, One Parkview Plaza, Suite 800, Oakbrook Terrace, IL 60181 USA, or contact Amy Lydic at +1-847-686-2236; fax, +1-847-686-2251; or email, inmm@inmm.org.

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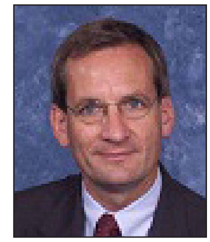


The Changing Nature of the Institute

By Larry Satkowiak
INMM President

I have been a member of the Institute of Nuclear Materials Management for more than twenty-five years. During that time I have seen many changes in the nature, structure, and size of the Institute as it has evolved to meet the needs of a changing world. Our nuclear materials management community has become more international in flavor with a broader diversity of interest areas both technical and non-technical. As an indication of the growth, I attended my first Annual Meeting in 1989 and there were fewer than 230 papers presented; we expect twice that number this year. In 2015, more than 35 percent of the presenters/attendees were from outside of the United States; it was only a fraction of that in 1989, another indicator of change.

Change is important and necessary to maintain relevance. Managing change is critical. Periodically, it is worthwhile to do an assessment to ensure that we are effectively addressing changes and are meeting the needs of our members. To that end, at the November meeting, the INMM Executive Committee agreed to engage an outside expert who specializes in developing strategies for building and sustaining successful professional societies. The assessment/strategic planning effort was kicked off in January with the membership survey that many of you participated in. In conjunction with our March Executive Committee and Technical Program Committee meetings, we held a day and a half strategic planning meeting to identify goals and objectives. The goals can be summarized as:

- Be recognized internationally as the leading professional society for the effective stewardship of nuclear materials and related technology
- Represent the breadth of the profession
- Strengthen the relationship between the policy and technical communities

Each of these broad goals had specific objectives developed and we are currently assessing implementation approaches to meet those objectives. Our intent is to use the results of this process to develop a stronger organization that meets the needs of our nuclear materials management community. I encourage all of the Institute's members to be engaged in this effort.

The INMM is well-positioned for the future. We continue to have a technical and policy relevance that provides the foundation of our mission. This relevance is important and will sustain us during difficult times and will be the engine for future growth. Our workshops provide a forum for extending the reach of INMM beyond the Annual Meeting. We will continue to encourage workshops as a way to demonstrate relevance to the international nuclear materials management community. We continue to be active internationally through our chapters and through partnerships with other related organizations such as the European Safeguards Research and Development Association (ESARDA), the World Institute for Nuclear Security (WINS), World Nuclear Transport Institute (WNTI), the International Atomic Energy Agency (IAEA), and the American Nuclear Society (ANS).

Looking Forward

The *18th International Symposium on the Packaging and Transportation of Radioactive Materials (PATRAM)* will be held September 18-23, 2016, at the Kobe Portopia Hotel in Kobe, Japan. PATRAM brings together experts from governments, industries, and research organizations worldwide to exchange information on all aspects of packaging and transport of radioactive materials around the globe.

December 5-9, 2016, the *IAEA International Conference on Nuclear Security: Commitments and Actions* will be held at the Vienna International Centre in Vienna, Austria. The INMM is a co-sponsor of the conference. INMM members are actively involved submitting abstracts and working on the Technical Program Committee.

Mission Statement

The INMM is an international professional society dedicated to development and promulgation of practices for the safe, secure and effective stewardship of nuclear materials through the advancement of scientific knowledge, technical skills, policy dialogue, and enhancement of professional capabilities.

Finally, from January 10-12, 2017, the *32nd Spent Fuel Seminar* will be held at the Washington Marriott Georgetown in Washington, DC, USA. Sponsored by the INMM Packaging, Transportation, and Disposition Technical Division in partnership with U.S. Nuclear Infrastructure Council, this annual event specifically targets key issues in spent fuel management and is international in scope.

Final Thought

During the past year, four additional student chapters were formed. The new chapters are Oregon State University (USA), Ahmadu Bello University (Nigeria), University of Florida (USA) and University of Utah (USA). We now have twenty-one student chapters and thirty-eight chapters overall! Several (as in seven to ten) years ago the Executive Committee made a concerted effort to engage the "next generation" of nuclear material management practitioners. Twenty-one student chapters later, I think we call the effort a success. This bodes well for the future of the Institute.

Dealing with Spent Nuclear Fuel

By Markko Koskelo
JNMM *Technical Editor*

The INMM organizes a Spent Fuel Management Seminar in January of each year. This year the seminar, the thirty-first in the sequence of these events, was held January 11-13, 2016, at the Washington Marriott Georgetown in Washington, DC, USA. The participants included representatives from several countries, multiple U.S. government departments and agencies, as well as participants from nuclear utilities, vendors, and consultants in this field. There is no formal requirement for the presenters to publish full papers for proceedings of this important seminar. However, we worked with Jeff England, chair of the Packaging, Transportation, and Disposition Technical Division, who organizes the Spent Fuel Seminar, to have a sampling of the presenters submit the content of their talks. We are happy to include three of the presentations as full papers in this issue. The papers were peer reviewed for clarity of the content and for adherence to the *JNMM* publication format and guidelines and give a very nice status update on the efforts to deal with spent nuclear fuel in three different countries.

The papers include a status of the Swedish spent fuel management program by Anders Sjöland, a report of the current Spanish Spent Fuel Plan by Alejandro Palacio and David Garrido, and a look at U.S. spent fuel management policy by Carlyn Greene.

Please see the excellent summary of the seminar as a whole, also written by Carlyn Green, that we have included as the first paper in this issue.

The last paper of this issue, A New Method to Measure the $^{240}Pu_{eff}$ Mass and to Verify the Pu Fissile Content in FBR MOX Fuel Assemblies, by Howard O. Menlove and Vladimir Henzl, is unrelated to the Spent Fuel Seminar but continues our streak of providing cutting-edge papers on new measurement technologies for new, more accurate or more cost effective ways to make measurements to know where the nuclear materials are and how much nuclear material is present at any given time.

Other Highlights

Book Review Editor Mark Maeillo takes a look at *Longing for the Bomb: Oak Ridge and Atomic Nostalgia,* by Lindsay A. Freeman. L. David Lambert, former chair of the INMM Nuclear Security and Physical Protection Technical Division, was invited to write the Taking the Long View column on the view from the international community.

Submit Your Paper

JNMM is always looking for interesting papers to publish in upcoming issues. See our author submission guidelines on page 56 of this issue and consider submitting an article for peer review and possible publication in the *Journal*.

JNMM *Technical Editor Markku Koskelo can be reached at mkoskelo@aquilagroup.com*

Spent Fuel Management Issues Highlighted at 31st Spent Fuel Seminar

By Carlyn Greene
The Ux Consulting Company, LLC, Roswell, Georgia USA

In mid-January 2016, about 100 nuclear industry professionals met in Washington, D.C., USA, for the 31st Annual Spent Fuel Management Seminar, sponsored by the Institute of Nuclear Materials Management (INMM), in partnership with the U.S. Nuclear Infrastructure Council (NIC). Participants and presenters included representatives from Japan, Korea, Spain, Sweden, the U.S. Department of Energy (DOE), U.S. national laboratories, the U.S. Nuclear Regulatory Commission (NRC), the Electric Power Research Institute (EPRI), utilities, cask vendors, consultants, and more. As always, the conference included a number of relevant presentations on a wide variety of topics related to spent fuel management, with consent-based siting for storage and disposal facilities, interim consolidated storage, the transportation of radioactive waste, and national spent fuel management programs being a few of the main focus areas. A few of the presentations on these topics are highlighted below.

Consent-based Siting

John Kotek, Acting Assistant Secretary for Nuclear Energy at DOE, provided a keynote address, in which he said the DOE is moving forward with plans for a consent-based siting process for the transportation, storage, and disposal of spent fuel and high-level radioactive waste (HLW). He emphasized that the Obama Administration believes nuclear power will continue to play a role in the U.S. clean energy plan and noted nuclear's role in the nation's defense activities as well. Kotek mentioned DOE is seeking public feedback on how a consent-based process should be designed. DOE began this consultation with a notice in the December 23, 2015, *Federal Register*, and has scheduled several meetings in various locations around the country during the first half of 2016 to obtain input on the process.

DOE continues to "decline to define" consent, he said, but expects that consent will include mutual terms of agreement between a community and the federal government. DOE will seek to address potential issues "early in the process" and "collaboratively." DOE will conduct its consent-based siting process in several phases, with each phase building on the last. The first phase, he said, is listening, and DOE wants to hear from a broad cross-section of citizens. The next step will be documenting the comments, then working with interested communities. DOE is working on getting feedback for at least the first six to eight months.

This strategy, Kotek said, reflects DOE's judgment that the one-size-fits-all approach used in the past is not feasible. In accordance with its January 2013 strategy, DOE will pursue a "multiple-path" approach that includes a pilot interim storage facility, a full-scale storage facility, a repository for commercial spent fuel, a separate repository for DOE-owned spent fuel and high-level waste (HLW), and deep borehole disposal for certain types of waste. He noted that on January 5, 2016, DOE announced that it has selected a team led by Battelle Memorial Institute to conduct a deep borehole field test in North Dakota.

Kotek said in response to a question that DOE "looks forward" to receiving input from the states and communities in which private industry already has obtained consent to build a consolidated interim storage facility (Texas and New Mexico). DOE has engaged with those communities and the companies involved to learn more about the proposals, and noted that before DOE could move forward with using a private facility, more authorization or implementing legislation would be needed. Of course, legislation is also required before DOE could work with any specific site for any new facility.

Llewellyn King, executive producer, *White House Chronicle*, spoke about consent-based approaches to siting nuclear waste facilities. King began his remarks by stating that when the Blue Ribbon Commission on America's Nuclear Future (BRC) recommended a consent-based process, he thought to himself, "Oh my God, here we go again" — a statement that resulted in a great deal of laughter. King acknowledged that consent is a good intention, but within a democratic country it is an assumed state. Nothing is done without consent. When a technical issue such as a nuclear waste repository is taken to the public, "Every garden club will get a lawyer and you're off to the races. You won't get it done — you won't get consent." Those people who are passionately opposed to nuclear will use the process to advance their agenda. Nuclear waste has been "God's gift" to the anti-nuclear movement, King remarked, because it is the easiest issue with which to alarm the public. The nuclear industry will continue to have perception problems until the open issue of nuclear waste is resolved.

The nuclear industry needs certainty, needs a waste repository, and needs progress toward having a repository in operation. King wishes the public would look at the waste issue in the context of human endeavors, as there is no other human endeavor that looks 300 to 1,000 years out (or a million years). King reasoned that if mankind continues to pollute the oceans the way we are now, the oceans will be dead in 300 years, yet the anti-nuclear establishment has convinced much of the public that the very small amount of nuclear waste that might still exist in 300 years is the only threat to humanity.

King was critical of President Obama's thesis about Yucca Mountain, since the local community gave its consent, but the state did not. "Where does consent begin and end? Does it lie with a single powerful politician?"

King concluded by emphasizing we have a "moral duty" to solve the nuclear problem, and maybe if we do, we can save the earth from being flooded as a result of climate change.

Consolidated Storage Facilities

In 2016, two applications from private entities for consolidated interim storage facilities (CISFs) in the United States are expected to be submitted to the NRC for review. Waste Control Specialists (WCS), working with AREVA TN and NAC International, has notified the NRC that it will submit an application in April for a CISF to be built at the WCS site in Andrews County, Texas. Holtec International, working with the Eddy-Lea Energy Alliance (ELEA), has notified the NRC it will submit an application by June for a CISF to be built on 1,000 acres in an undeveloped and isolated area in southeast New Mexico that ELEA owns. Both applicants emphasized that CISFs do not replace the need for a permanent disposal facility, but they would provide several benefits to the industry, some of which include:

- Promoting and supplementing the development of a permanent repository
- Facilitating the federal government's removal of spent fuel and reactor-related greater-than Class C (GTCC) waste from permanently shut-down plants earlier than anticipated with a repository-only approach
- Demonstrating competency in the consent-based process, in the licensing process, and in the transportation of spent fuel
- Both companies claim their facilities could be ready to be operational as early as 2020.

Both WCS and Holtec envision that the first spent fuel to be shipped to their sites will be casks from permanently shut-down sites, as envisioned by the Blue Ribbon Commission (BRC) on America's Nuclear Future, both claim broad support for their proposed facilities, and both are seeking a forty-year license.

The WCS facility will be sized to store up to 40,000 metric tons (MT) of spent fuel and reactor-related GTCC waste. It will be built in phases, with the first phase to accept approximately 5,000 MT of spent fuel from permanently shut-down sites. To the extent possible, the facility will use existing licensed canisters from both AREVA TN and NAC International. The initial license application for approximately 5,000 MT of spent fuel will cover about 80 percent of all spent fuel and GTCC waste at stranded sites. Other sites that could send spent fuel and GTCC waste to WCS in the future, and the storage systems employed at those sites, will be addressed through future amendments to the site-specific license as needed.

Holtec plans to deploy the HI-STORM UMAX (Underground, Maximum Capacity) system at its facility. Holtec has said that the facility will be designed to store "the entire complement of canisters currently deployed at independent spent fuel storage installations (ISFSIs) around the country." At a later date via license amendments, Holtec plans to expand the contents list of the HI-STORM UMAX Certificate of Compliance (CoC) to incorporate all Holtec canisters, and all canisters from shutdown plants, regardless of the type of canister or vendor a particular site currently employs. The facility will eventually have a capacity of 4,000 canisters, although the initial application will seek approval to store 500 canisters of "priority waste," which is defined as canisters from permanently shut-down/decommissioned sites.

Transportation

"Without transportation there is no nuclear industry," said Henry-Jacques Neau, Secretary General of the World Nuclear Transport Institute (WNTI), who offered one of several presentations related to transportation of radioactive materials. Radioactive materials are safely transported by sea, air, road, and rail, but the past impeccable safety record does not guarantee an equally safe record in the future unless the industry remains diligent. Transport costs are a very small but crucial part of the industry. Neau cautioned against cutting corners when arranging for transportation of radioactive materials, and urged companies who need transportation to use experienced and reliable transport companies. He also encouraged all companies who need to transport radioactive material to join WNTI in order to participate in developing regulations.

Energy*Solutions*' Robert Quinn also highlighted "70 Years of Transportation Safety and Counting" in his presentation. Energy*Solutions* operates a large fleet of Type A and B truck casks and more than 350 railcars. Although the majority of Energy*Solutions*' shipments are by rail, its truck fleet alone logs more than 8 million miles per year and transports more than 300 radioactive shipments per month. In the United States, more than three million packages of radioactive materials are shipped annually — equating to an average of more than 8,200 per day.

Quinn detailed the extensive global transportation safety record, including that of the U.S. Navy, which has completed about 850 shipments of spent fuel, totaling over 1.6 million miles of transport. More than 1,300 shipments of commercial spent fuel have been safely completed since the mid-1970s, plus sixty shipments of foreign research reactor fuel that have been shipped to and within the U.S. by sea, land, and air. Internationally, more than 70,000 MT of spent fuel has been transported by road, rail, and sea within and among the UK, France, Germany, Sweden, Japan, and other nations. None of these shipments has resulted in a failure of a package; therefore, no release of radioactive materials has occurred during any of these spent fuel shipments.

Both Quinn and Neau emphasized that the regulations that govern nuclear material transportation are comprehensive and ensure its safety. Tests and evaluations have shown that transportation packages can survive severe "real world" accident scenarios and fire scenarios. The NRC recently issued two draft reports for public comments related to the evaluation of transportation packages in a severe fire. Neau noted, however, that regulatory harmonization is a challenge, as different countries have different regulations or requirements.

Quinn pointed out the somewhat specious concerns about the ability to handle and manage the transport of spent fuel from reactor sites to an interim storage or permanent disposal facility. The DOE has stated that it would use a "mostly rail" approach, and would ship about 3,000 MT per year. Using current rail casks, this would result in the shipment of about 300 packages per year. If a dedicated train is used for the shipments, three packages would likely be on each shipment, so that would be 100 shipments per year, or about two shipments per week. This represents only 0.01 percent of the total radioactive waste packages already being shipped in the U.S. each year. This would not be an "overwhelming challenge, and in fact, represents only a minimal increase in the annual shipments of radioactive materials."

Sven Bader of AREVA Federal Services, LLC, noted that AREVA safely completes around 4,000 shipments of radioactive waste globally each year, including more than 200 shipments of spent fuel and of vitrified and compacted waste, more than 150 shipments of mixed-oxide (MOX) fuel, more than 300 shipments of low-level waste, more than 2,700 front-end shipments, around 150 shipments for research reactors and laboratories, and more than 400 shipments of heavy industrial equipment. "Transportation of used nuclear fuel is a well-understood and demonstrated activity," he said. In the U.S., AREVA has completed more than 250 shipments of foreign research reactor spent fuel to Savannah River and Idaho, as well as shipments of spent fuel in the 1980s from Dresden and Oyster Creek to West Valley, and more.

Bader noted that AREVA has a large fleet of casks dedicated to spent fuel and vitrified waste, along with trucks, railcars, and trailers that are designed for radioactive waste transportation. At the Valognes terminal in France where packages are transferred from railcar to truck, more than 1,000 such transfers occur each year. This terminal, which is 40 kilometers from the La Hague reprocessing facility, is the link between the La Hague facility and the railway network.

AREVA TN has obtained a license for a universal transport package that can transport nine different canister types that contain high burnup spent fuel. The MP197HB transportation package is currently being fabricated at Hitachi-Zosen and will be ready for deployment in 2016. It is approved for use by truck, rail, or marine transport. AREVA TN also has a conceptual design of a transportation package for uncanistered spent fuel that would be a reusable transportation package for sixty-one boiling water reactor (BWR) fuel assemblies, or twenty-four pressurized water reactor (PWR) fuel assemblies. It would be designed to handle short-cooled, high burnup fuel, and damaged spent fuel in damaged fuel cans. This transportation package would take spent fuel directly from the spent fuel pool.

An AREVA-led team is designing a railcar under a contract with DOE that will meet the Association of American Railroads (AAR) standard S-2043. An AREVA-led team is also developing an initial "site-specific de-inventory implementation plan," which would provide information on potential de-inventory activities at select sites with shutdown reactors. The project involves identifying tasks and interfaces necessary to perform these activities, including near-site transportation infrastructure and routes to railroads or spurs, and other activities. Bader noted that significant effort might be required to move the spent fuel from the "stranded" ISFSIs where nothing remains at the site except for the ISFSI due to loss of infrastructure at the sites.

Finally, Bader reviewed the "La Hague Advantage" where about 3,000 fuel assemblies are shipped each year. La Hague is capable of unloading casks in both dry (250 casks per year) and wet (130 casks per year) facilities. Between January 2000 and June 2014, La Hague has received 3,363 casks, which included 1,080 damaged and 229 failed fuel assemblies. Unloading has included short-cooled, high burnup spent fuel assemblies.

Michele Sampson, chief, spent fuel licensing branch in the NRC's Division of Spent Fuel Management, provided a regulatory perspective of transportation. Sampson reviewed some of the agency's accomplishments over the last year, including several works-in-progress, and she offered a view of what the NRC is expecting to accomplish in 2016. She touched on fuel retrievability, noting that the current guidance on retrievability reflected the expectation that a permanent repository would be available in the near term, and that the current guidance may have "unintended consequences" including increased worker dose and degradation of the confinement boundary. "Ready retrieval" is defined as the "ability to safely remove, with no operational safety problems, the spent fuel from storage for further processing or disposal."

In July 2015, the NRC held a public meeting on fuel retrievability, and the NRC issued a draft Interim Staff Guidance (ISG), Revision 2, on the issue, then held another public meeting in October. The draft ISG focuses on the safety and design bases to allow maximum flexibility for an undefined storage duration. In March 2015, the NRC issued a draft Regulatory Issue Summary (RIS), *Considerations in Licensing High Burnup Spent Fuel in Storage and Transportation,* and in May 2015 issued a NUREG/CR report *Mechanical Fatigue Testing of High-Burnup Fuel for Transportation Applications*.

Coming up this year, some of the NRC's activities include comparing 10 CFR Part 71 to the International Atomic Energy Agency's (IAEA's) 2012 edition of *Regulations for the Safe Transport of Radioactive Material*, and will consider the draft IAEA Safety Guide, *Package Design Safety Report*. The NRC will continue work on licensing program improvements, including:

- Compatibility of storage and transportation regulatory framework; and
- Regulating ISFSIs at decommissioned reactor sites, which will include reviewing the framework for unloading capability at ISFSIs.

Sampson presented a few licensee considerations when looking forward to the future transport to a storage facility. These items include:

- Does the transport package meet the CoC;
- Will any approvals from the NRC be needed – if so, the licensee should keep in mind the time required to prepare an application for submittal to the NRC and allow for a review time of one to two years;
- Understand that the NRC could have concurrent applications to review prior to a large-scale transportation campaign;
- Ensure that records to support shipments are maintained;
- Ensure that the records are transferred to the away-from-reactor storage facility to support future transports for disposal.

Mark Lombard, acting deputy director of the NRC's Office of Nuclear Material Safety and Safeguards, noted in his presentation that in 2015, the agency completed the review of fifty-one transportation cases. He also pointed out that the NRC completed the analyses and reviews of the MacArthur Maze and Newhall Pass severe fire studies.

The DOE is preparing for a large-scale transportation program to ship fuel from reactor sites to one or more interim storage facilities. Matt Feldman of Oak Ridge National Laboratory (ORNL) informed participants of DOE's transportation system development undertaken by the Nuclear Fuels Storage and Transportation Planning Project (NFST), which was established in 2013 to plan for interim storage and transportation. NFST activities align with BRC recommendations and with DOE's 2013 *Strategy for the Management and Disposal of Used Nuclear Fuel and High-level Radioactive Waste* (the Strategy), and are consistent with both existing statutory authorities and with budget direction and authorization. One of NFST's high-level objectives is to "prepare for the large-scale transportation of spent nuclear fuel and high-level waste, with an initial focus on removing spent fuel from the shutdown reactor sites."

The transportation system is organized into three primary elements: institutional, operational, and hardware. A transportation planning framework (TPF) establishes the basis for stakeholder collaboration. It is a living document that will be developed and modified as needed throughout the life of the transportation system. NFST is addressing the development of a standardized route selection process, and will work with stakeholders to finalize the route selection process so that all interested parties will understand how routes are chosen.

DOE is looking at what activities will be needed to remove spent fuel from shutdown sites. Plans cover all aspects of the spent fuel removal from permanently shutdown sites, and will be developed for all shutdown sites. Plans are complete for Connecticut Yankee and Big Rock Point, and a plan for Humboldt Bay is being developed. Additional site plans will be developed as funding permits, with two more site plans being developed in Fiscal Year 2016, although the sites are still to be determined.

The order of planned completion does not indicate a preferred order of servicing, and DOE has not made the decision yet to take spent fuel from the shutdown sites first, Feldman emphasized.

Patrick Schwab of DOE's Office of Nuclear Energy (DOE-NE) presented the ATLAS Railcar Project. ATLAS is not an acronym, but the program was named after the mythological Greek titan who held up the sky. DOE can work on developing a railcar program since this program is "destination independent" and will be necessary regardless of where the material is eventually transported. DOE has agreements with two railroad companies that state all cars will comply with the Association of American Railroads (AAR) Construction Standards. Testing and approval of railcars is a long lead time project, and DOE expects prototype fabrication and delivery to occur in 2017 into 2019, with the ATLAS project scheduled to be complete in 2022. DOE signed a forty-three-month, $8.63 million contract with AREVA Federal Services on August 21, 2015. More procurements will be needed to finish the ATLAS project. The ATLAS railcar will carry commercial spent fuel casks weighing up to 156 tons.

Overview of U.S. NRC Work in 2015 and Looking Forward

Officials from the NRC offered a review of some of the accomplishments in the past year, and a preview of plans for the current year, along with some external factors that affect the agency's work.

Glenn Tracy, deputy executive director for operations, led the way as he pointed out that in Fiscal Year (FY) 2015 (which began on October 1, 2014), the NRC approved the renewed license for the Calvert Cliffs independent spent fuel storage installation (ISFSI) and issued a renewed license, completed the safety review for the Prairie Island ISFSI (and issued that renewed license in December 2015), continued the technical review of the VSC-24 cask system renewal application, completed a draft of Revision 1 of the *Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel* (NUREG-1927, Rev. 1), completed the Safety Evaluation Report (SER) for the Yucca Mountain license application, harmonized 10 CFR Part 71 with IAEA SSR-6 (Regulations for the Safe Transport of Radioactive Materials), and worked with Oak Ridge National Laboratory (ORNL) on high burnup fuel testing, Phase 1.

In FY 2016, the staff will continue to develop a risk-informed regulatory framework with a target of finalizing the framework in summer 2016, will continue to work with ORNL on the high burnup fuel tests, Phase 1, and will complete the supplement to the Yucca Mountain Environmental Impact Statement (EIS).

External factors that could impact the NRC's work include the submission to the NRC of two applications for consolidated interim storage facilities (CISFs), a possible topical safety analysis report (TSAR) from DOE on a federal consolidated storage facility, possible legislative actions, and transportation package applications associated with interim storage facilities that could be submitted for review and approval. The NRC coordinates work with DOE through regular meetings on technical topics and research and development activities. The NRC could provide independent regulatory support for DOE's consent-based siting initiative.

Mark Lombard returned to the podium when he reviewed the "ghosts of spent fuel past, present, and future," as he cited numerous achievements the staff realized in 2015, including the review of fifty-one transportation casks and nineteen storage cases, plus a number of other tasks that were completed and reports issued, in both final and draft form. The staff is anticipating one storage renewal application to be submitted in 2016, two in 2017, one each in 2018 and 2019, then a surge of renewal applications to be submitted in 2020 (six applications), followed by a steady stream of renewals through 2024.

Al Csontos, chief of the Renewals and Materials branch, emphasized that the key to managing the workload is having an updated guidance infrastructure, which the staff has been working to implement, in collaboration with industry. Staff is taking an "operations-focused approach for renewals and aging management that is learning, proactive, and responsive," and which is characterized by the following:

- Based on achievable operational methodologies
- Condition based monitoring and/or in service inspections (ISI) that are based on technically defensible criteria
- Assessment of monitoring and ISI findings and data
- Criteria for action (no action, repair, replace, other mitigation)
- Aggregate, trend, and report operating experience (OpE)
- Learning aging management programs (AMPs) that consider and respond to OpE.

The updated guidance infrastructure is expected to result in a stable, predictable, reliable, and consistent NRC review process along with clear, open, and transparent regulatory expectations, all of which will culminate in an efficient review process for the renewal applications.

Renewal guidance for the staff includes NUREG-1927 Rev. 1, the Managing Aging Processes for Storage (MAPS) report, inspector guidance of licensees' aging management activities, and a Regulatory Guide that discusses guidance framework and is a vehicle for the potential endorsement of industry guidance.

Others outside the NRC are also developing renewal infrastructure, including: Consensus ASME Boiler and Pressure Vessel Code Section XI Code Case for ISI of dry cask storage canisters; Consensus ACI Guide for ISI of concrete storage overpacks; NEI 14-03, Rev.1 Format, Content & Implementation Guidance for Dry Cask Storage Operations-Based Aging Management and Development of in-situ NDE capabilities.

The draft Standard Review Plan was published for public comment in July 2015. Staff is currently addressing the comments received, and will finalize the report. Staff will engage with the Advisory Committee on Reactor Safeguards (ACRS) in March and April of this year, then publish the final guidance in summer 2016. All the while, staff will continue stakeholder engagement with public meetings, and with the Nuclear Energy Institute (NEI) on its proposed guidance, NEI 14-03. For the MAPS report, staff will engage with the ACRS in May 2016, and publish the draft document for public comment in June, with the final guidance ready in summer 2017.

Overview of EPRI Work in 2015 and Looking Forward

Keith Waldrop, principal technical leader at the Electric Power Research Institute (EPRI), noted that EPRI is working to develop aging management guidance for the inspection and mitigation of welded stainless steel canisters, and to develop and demonstrate the inspection capability of these canisters. EPRI has performed a Failure Modes and Effect Analysis (FMEA) to systematically identify the potential failure modes of a cask system, the relative likelihood of the identified potential failure modes, and consequences to the storage system. The FMEA's scope is limited to welded stainless steel canisters exposed to air during the lifetime of extended storage, and it is focused on identifying credible failure modes for welded stainless steel canisters under design basis conditions. Through the FMEA process, chloride-induced stress corrosion cracking (CISCC) has been identified as the most likely degradation mechanism to lead to through-wall penetration of welded stainless steel canisters among various active degradation mechanisms. The degraded canister would release over-pressured helium and then allow air ingress over time.

EPRI is working with an advisory panel consisting of EPRI members, cask vendors, and national laboratories to review and revise drafts of the CISCC aging management guidelines, and also is working with the NEI task group on NEI 14-03 (Format, Content and Implementation Guidance for Dry Cask Storage Operations-Based Aging Management). Publication of the guidelines is scheduled for later in 2016. A key contributor to the guidelines is a susceptibility assessment published in 2015 that provides a systematic, scientifically based method to prioritize canisters for inspections.

EPRI is also developing non-destructive evaluation (NDE) techniques for use in canister inspections, which is a challenge because of the high radiation dose, high temperatures, and difficulty in accessing the canister. EPRI is building canister mockups to demonstrate NDE capability with a goal of having an in-situ demonstration of an NDE probe in combination with a delivery system in 2017. EPRI encourages the demonstration of multiple technologies by multiple stakeholders. EPRI has eight quality assurance (QA) flaw mockups manufactured for industry use, which have been used by two vendors and two universities. Sandia National Laboratories also sent EPRI a full diameter, partial length mockup for continued development.

NDE techniques that are under development by EPRI include Eddy Current Array (ECA), guided waves to find defects in inaccessible areas, and acoustic emission, which has the potential to monitor the canister from outside the cask. Vendors are actively working in several areas including visual techniques and ultrasonic techniques for defect sizing.

In addition, EPRI is developing two robotic delivery systems to improve inspection capabilities — a magnetic wheel robot and a vacuum suction robot.

Waldrop also discussed the High Burnup Spent Fuel Data Project, for which EPRI is the lead. This project is intended to provide data on the behavior of multiple types of high burnup cladding under typical dry storage conditions, and to provide data for benchmarking models to predict the performance of high burnup fuel over extended time periods. The Aging Management Programs for high burnup fuel in the Calvert Cliffs and Prairie Island ISFSI license renewals rely on this research project. Other storage renewals under review or planned will also benefit from this research.

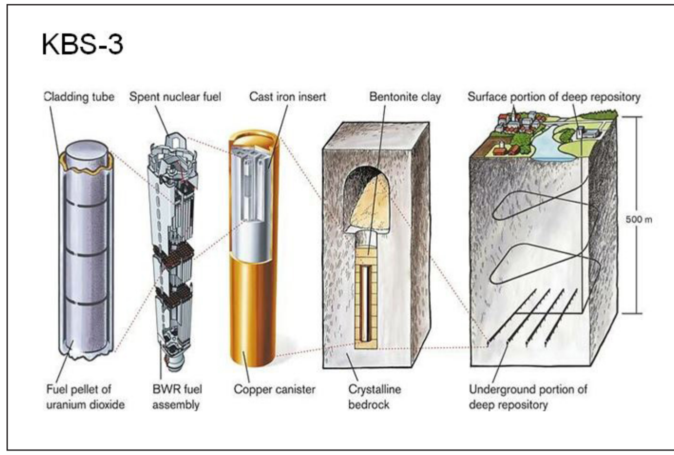
The High Burnup Spent Fuel Data Project contains several key activities: identify, extract, and ship sister rods to a national lab for baseline properties; submit and obtain an NRC license amendment to store the cask; modify the cask and fabricate instrumentation; load the cask; and collect data (temperatures, gas composition). This project started in 2013 and is scheduled to load the cask and begin collecting data in 2017. Monitoring will continue through at least 2027, at which time the cask will be shipped to an examination facility for additional analysis and comparison to baseline measurements.

National Programs

Anders Sjöland, of the Swedish Nuclear Fuel and Waste Management Company (SKB), discussed the Swedish nuclear fuel management program.

Sweden's nuclear reactors generate about 45 percent of the country's electricity and are expected to generate approximately 12,000 MT of spent fuel. Sweden is building a final repository for short-lived radioactive waste (SFR), and already has a central interim storage facility for spent nuclear fuel (Clab). Construction on the SFR is expected to begin in 2019 and operations are expected to begin in 2023. The SFR will have a capacity of 130,000 cubic meters, of which 50,000 cubic meters can be intermediate-level waste.

Figure 1. KBS-3 concept



SKB recently submitted an application to increase Clab's storage capacity from 8,000 tons to 11,000 tons, as the 8,000 ton capacity is expected to be reached in 2022. This capacity increase will be reached primarily through the use of higher density spent fuel pool racks, and removal of non-fuel items. A third pool could be added at Clab, but that is not included in the present application. The figure above, from Sjöland's presentation, illustrates the KBS-3 concept.

For the permanent repository for spent nuclear fuel, a license application for a spent fuel repository at Forsmark and an encapsulation plant in Oskarshamn was submitted in March 2011. The facility will accept the spent fuel that is currently stored in Clab, as well as spent fuel that will arise from the currently operating reactors. Final disposal will be according to the KBS-3 method, which was developed by SKB (and adopted by Posiva in Finland), with vertical placement of the canisters. The design capacity is for 6,000 canisters, which corresponds to 12,000 MT of spent fuel. It will be licensed to operate for 60 years. The KBS-3 method is shown in the accompanying figure, which was included in Sjöland's presentation.

SKB needs five approvals before construction can begin — the Swedish Radiation Safety Authority (SSM), the Environmental Court, the communities of Östhammar and Oskarshamn, and the Swedish government. The Environmental Court announced in December 2015 that it will declare the application complete. SSM is expected to also declare the application complete, after which time the formal review can begin. SKB hopes that the Swedish government can issue a construction permit in 2018, construction could then take place between 2020 and 2030, with operations beginning in 2030. Read Sjoland's presentation on page 14 in this issue of the *Journal*.

Jean-Michel Hoorelbek of Andra, the French radioactive waste management agency, noted that France recognizes spent fuel as a resource — not as waste. Any future placement of spent fuel in a repository will be fully reversible, as France plans to store spent fuel in a repository until the fourth generation fast breeder reactors, which will use the plutonium and depleted uranium, are built. The Cigéo geological disposal facility project is designed for the waste generated by existing nuclear facilities, to include vitrified HLW that is the result of reprocessing spent fuel, and intermediate level waste (ILW). The total volume of vitrified HLW to be disposed of is 10,000 cubic meters, and the volume of ILW to be disposed of is 73,000 cubic meters. The repository must allow for reversibility, which means that the equipment, waste packages, and disposal cells are designed to allow package retrieval for at least 100 years after emplacement to provide future generations with management options. During operations of the repository, research and development and monitoring will inform future decisions.

Public debate was conducted in 2013. This debate resulted in several changes to the project, which will now include an industrial pilot phase that will be integrated into the facility startup phase. A master plan for Cigéo operation and closure will be drawn up and regularly revisited, and the public will be more involved than in the past. The pilot phase will be followed by full facility operation. The application for the Cigéo repository is due to be submitted to the French regulator in 2018, with a license expected in 2021. The updated provisional schedule calls for the pilot phase to start in 2025 with inactive tests being conducted from 2025 until 2029, with emplacement of the first waste packages in 2030. A gradual increase in activity will take place from 2030 to 2035 so the emplacement progresses gradually until the nominal disposal rate is achieved. The Cigéo facility can be adapted to accommodate the direct disposal of spent fuel if it becomes necessary.

In Japan, Masumi Wataru of the Central Research Institute of the Electric Power Industry (CRIEPI) noted that the new basic energy plan has nuclear power as an important baseload source of electricity and the nuclear fuel cycle will continue to be promoted. To that end, Japan's reactors are restarting incrementally, with a goal of having nuclear power contributing more than 20 percent of Japan's electricity needs by 2030. The government will lead the efforts for the final disposal of HLW through the repository selection process. Electric power companies aim to secure additional spent fuel storage capacity of approximately 6,000 MT, which will include 4,000 MT of additional capacity by 2020 through currently planned measures, and another 2,000 MT by 2030. Japan is testing the life-limiting characteristics of metal and concrete casks. The key concern of concrete cask performance is the salt related stress corrosion cracking of stainless steel; for metal casks the key performance issue is seal integrity.

Professor Il Soon Hwang presented the Korean spent fuel management program. Korea imports 97 percent of its energy, but one-third of the country's electric power is from nuclear plants. Twenty-four units are currently in operation, four are under construction, and six are being planned, as Korea aims to reduce carbon emissions by 37 percent by 2030. Hwang said that while Korea's nuclear program has been a success, its spent fuel management program has been a failure. Each time a disposal site has been named, riots erupted, and the site was abandoned. Nuclear plants have been expanding storage capacity since 1990 by re-racking the pools from low-density racks to high-density racks, and moving spent fuel from an older unit to a younger unit at the same site. The roadmap set forth by the Public Engagement Commission on Spent Fuel Management (PECOS) calls for an interim storage site to be selected using a consent-based process by 2020, the site to be operational by 2024, and a high-level waste repository to be operational by 2051.

Decommissioning

Edward Davis, president of the Pegasus Group, presented a global decommissioning picture. Davis noted that the International Energy Agency (IEA) estimates that over the next twenty years "and beyond," more than 200 nuclear plants, which generate 150 gigawatts of electricity, are expected to be retired. To date, he said, over 157 nuclear plants worldwide have been shut down and/or are undergoing decommissioning. The main drivers for the plant retirements are as follows:

- 75 percent have achieved their expected economic lifetime
- 20 percent closed prematurely for political or regulatory reasons
- 5 percent closed after an accident.

The average age of the global nuclear fleet is twenty-seven years; in OECD (Organization for Economic Cooperation and Development) countries, which together account for about 80 percent of existing nuclear capacity, the average age of the reactor fleet is twenty-five years, but in non-OECD countries (excluding Russia), half of the nuclear capacity is less than fifteen years old. Currently, OECD countries are building twenty-one new reactors and non-OECD countries are building forty-six new reactors. Out of the world's 441 operating reactors, sixty-five are more than forty years old; 220 reactors are more than thirty years old; and 356 reactors are more than twenty years old.

Citing the World Nuclear Association (WNA), Davis noted that in 2015, ten new reactors began commercial operations, while eight reactors were permanently shut down. Through 2040, he said that about 150 GWe of nuclear capacity is expected to be retired, which is equivalent to 38 percent of the current installed capacity.

Most of the reactor retirements are in "mature" markets, with the average rate of retirements being about five GWe per year. The rate of reactor retirements accelerates in the first half of the 2020s as reactors built in the 1970s retire. Another period of accelerated retirements is expected in the 2030s, especially if life extensions in the U.S. are not renewed for an additional twenty years.

The U.S. nuclear fleet is the oldest in the world, with an average age of 33 years. Seventy-five U.S. reactors have received a twenty-year life extension. Without additional new units being brought online beyond those currently underway, the total U.S. installed capacity will begin to decline starting around 2027.

Davis noted that decommissioning cost estimates vary, but based on U.S. data, decommissioning cost estimates are in the range of $750 million to $1 billion per 1,000 megawatt.

According to Davis, the decommissioning market size is in the range of $100 to $150 billion through 2040 as follows:

- U.S. at $30 billion
- France at $25 billion
- Russia at $15 billion
- UK at $20 billion
- Germany at $30 billion

Table 1. International Energy Agency (IAEA) decommissioning costs through 2040

Table 11.2 ▷ Cumulative global investment and associated costs in nuclear power in the New Policies Scenario, 2014-2040 ($2013 billion)

	Investment in nuclear plants*	Associated costs		Total capacity additions (GW)
		Fuel cycle	Decommissioning	
China	345	191	-	132
European Union	301	220	51	45
United States	247	236	15	33
Korea	103	78	1	29
India	96	37	1	34
Japan	37	54	10	6
Rest of world	406	161	27	101
Total	**1 533**	**977**	**104**	**380**

* Investment in new plants and for uprates and refurbishments for life extensions at existing ones.

Davis noted that decommissioning costs can vary widely depending on the decommissioning approach selected, and specific requirements and regulations of a given country, as well as industry practices.

The conference contained a number of other valuable presentations, including updates about research on spent fuel storage casks during extended storage periods, more specific information about ongoing work at Idaho National Laboratory, DOE's deep borehole disposal program, cask vendor technology updates, additional details on aging management programs for spent fuel casks and ISFSIs, and more. The 32nd annual Spent Fuel Seminar will be held in Washington, DC, January 10-12, 2017; based on the success of the first thirty-one seminars, this is one conference not to be missed.

This paper is a summary of the presentation made at the 2016 INMM Spent Fuel Seminar.

Status of the Swedish Spent Fuel Management Program

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Abstract

Sweden has had a nuclear energy program since the 1950s. Beginning in the 1970s, Sweden and SKB began the development of a geological repository concept, which resulted in the present KBS-3 system. Forsmark was chosen in 2009 as the site for the repository, and a licence application to build the repository was submitted in 2011. The review process is ongoing, and is expected to result in the start of construction around 2019.

Introduction

Sweden has had a civil nuclear energy program since the 1950s. After an initial period of first generation reactors, a program consisting of twelve second generation reactors at four sites was established. These were put into operation between 1971 and 1985, and are of boiling water reactor (BWR) type and pressure water reactor (PWR) type. After the Three Mile Island accident in 1979, a referendum on the future of nuclear power in Sweden was held. The result was, after political negotiations, that all nuclear power should be discontinued in 2010. However, in 2004 a new compromise was made between the political parties, which resulted in the shutdown of the reactors at the Barsebäck site in southern Sweden, whereas the remaining ten could go on without any set time limit. It was also agreed that the existing reactors could be replaced, but that the total limit was ten reactors at three sites.

After the parliamentary elections in 2014, which resulted in a coalition between the social democratic party and the green party, without its own majority and which replaced the former center-right government, political decision resulting in increases in the special tax for nuclear power in Sweden, together with the dramatically decreasing electricity prices, led the largest owner of nuclear power in Sweden, the state-owned company Vattenfall, to make the decision to shut down two reactors (Ringhals 1 and 2), quickly followed by the decision by German-owned Eon, which is the majority holder of Oskarshamn 1-3, to make a similar decision regarding Oskarshamn 1 and 2 (Oskarshamn 2 will not be restarted and shall be changed to service operation with all nuclear fuel removed from the plant) by the end of 2016, according to current plans. It remains to be seen exactly what the outcome of these processes will be.

Early on, the Swedish spent fuel program was established. It was 'kick-started' after another election, in 1976, when the "conditional" law was established, which required the owners of nuclear power to "solve" the issue of safely taking care of the spent fuel as a condition for operating and starting new reactors. Swedish Nuclear Fuel and Waste Management Co. (SKB) was established after a time with a different name, and a substantial program on the research and development of a geological repository for spent fuel was commenced.

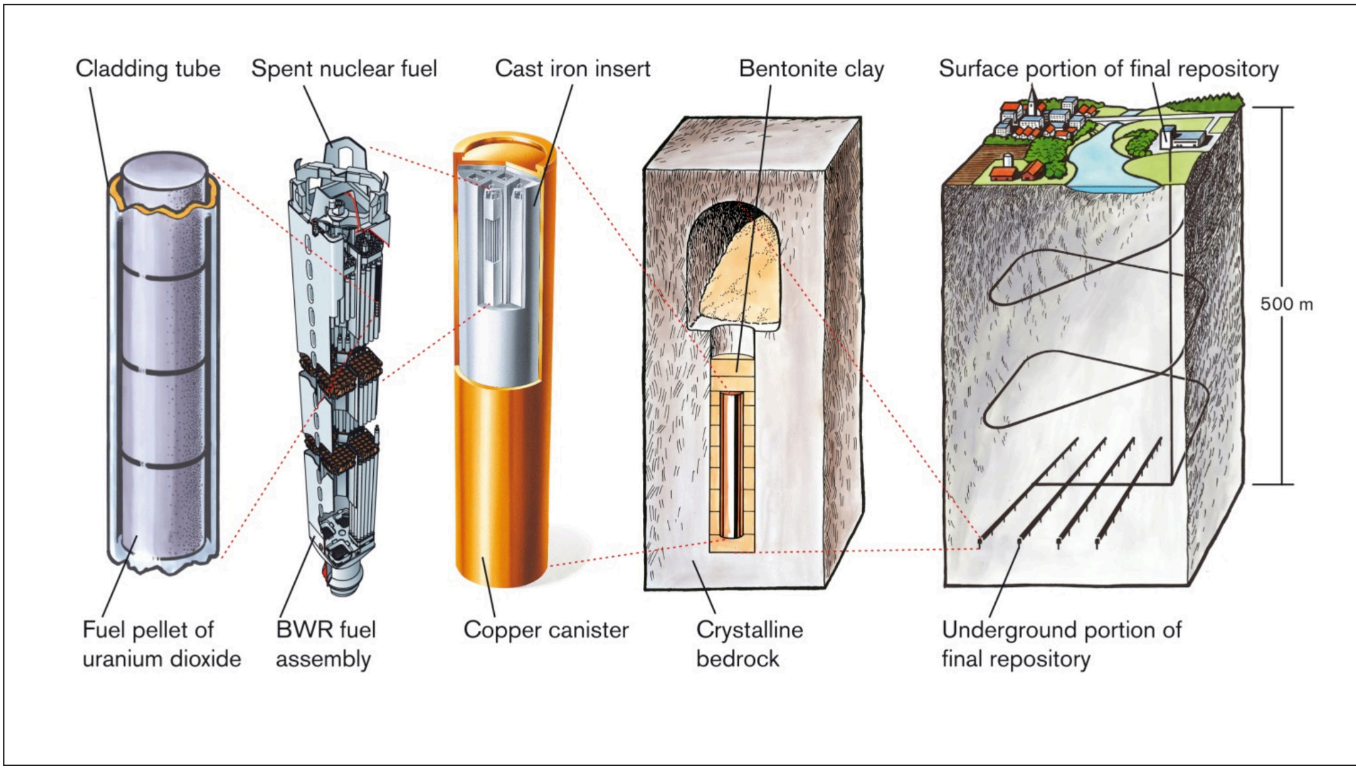
Swedish Geological Final Repository — the KBS-3 System

The KBS-3 system has been invented and developed by SKB, and adopted by Finnish Posiva. It is a multi-barrier system where the spent fuel is put into a canister consisting of an insert of cast iron, which is surrounded by a copper shell with a thickness of 50 mm (see Figure 1). The canister is deposited into the Swedish granitic bedrock at a depth of approximately 4,500 m, in deposition holes surrounded by the natural clay bentonite. All cavities in the bedrock are backfilled before closure of the repository. After closure the repository can be left without supervision, and no further action has to be taken by future generations.

Copper was chosen as the canister material because it is considered to have very low corrosion rates at anoxic conditions, which will be the case in the repository after a fairly short period after closure. The oxygen is consumed primarily by the microbes in the system.[1]

Bentonite was chosen because of its spectacular ability to absorb water and in doing so to swell and seal off the canister,

Figure 1. The KBS-3 geological repository concept



and thereby protect the canister from inflowing ground water. It also retains radionuclides in the case of breakage of the canister to some extent.[2, 3, 4]

The Swedish granitic bedrock was chosen because of its stability; the Swedish granitic bedrock is more than 900 million years old and at the chosen site at Forsmark, the bedrock is around 1.8 billion years. However, it is fractured and contains groundwater, which is rectified with the use of bentonite to protect the canister. The Swedish land area is by a vast majority covered with crystalline bedrock, so not many alternatives exist. All rock types have their pros and cons, and granite must be considered attractive by comparison.[5]

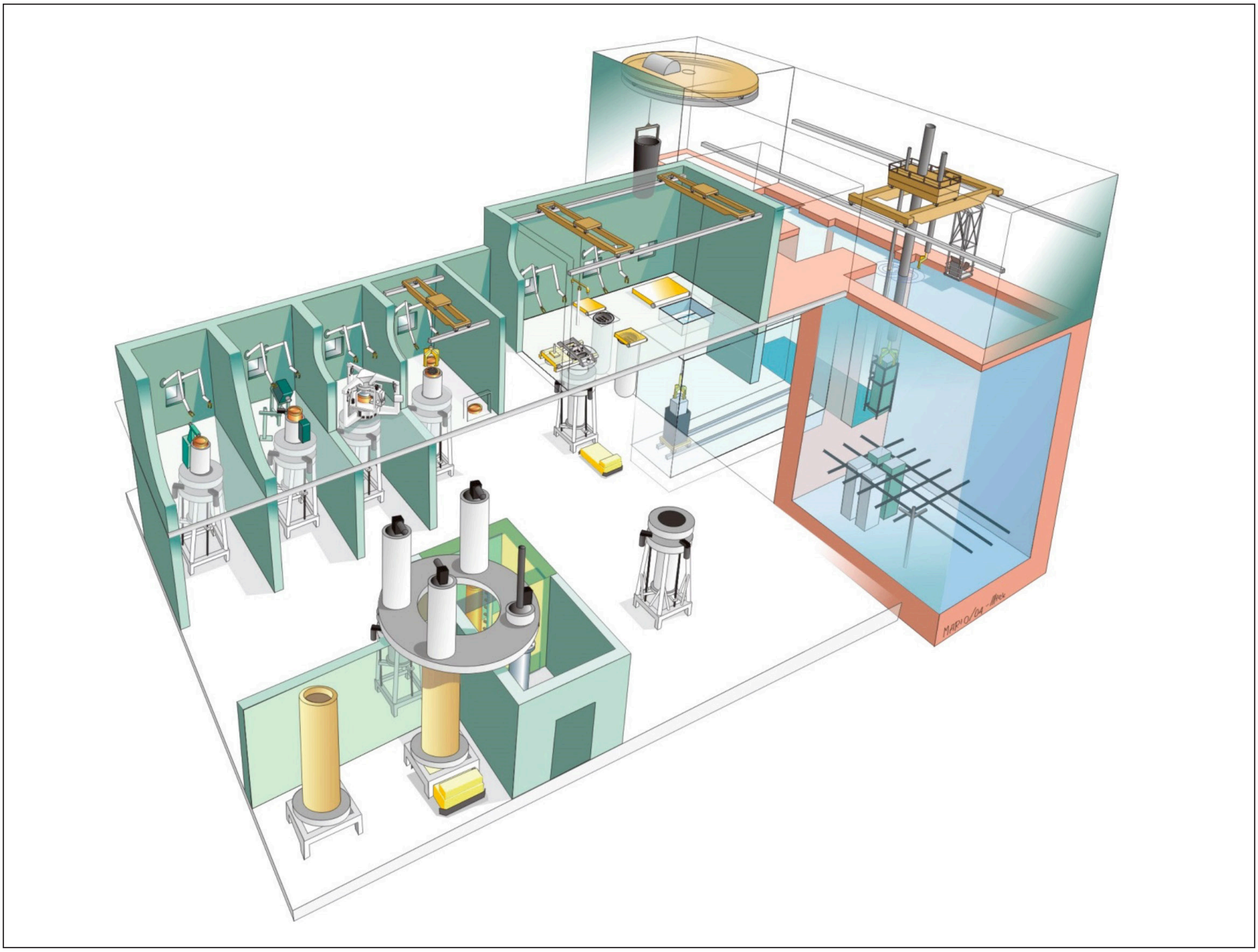
The KBS-3 system is highly coupled, as the bentonite has very high swelling pressures, typically several MPa, and the bentonite will push against the rock and canister tightly. It has been tested in full scale in the prototype test at Äspö HRL.[6]

KBS-3 is in principle retrievable, although not easily. SKB has demonstrated methods to retrieve deposited canister out of a closed geological repository. This is a lengthy and difficult process, but possible. It is a point in itself that it is so, as the primary function of the repository is that it should not be possible to take the spent fuel from the repository very easily. The reason the concept is in its nature retrievable is not that it is necessary to do anything with the repository in the future but so that future generations can decide to do what they like with the fuel, for example utilize the remaining energy in the fuel. Everything is done in order to analyse all consequences of the geological repository, but if something important would be found in the future it will indeed be possible to remedy the situation.[7, 8]

Encapsulation Plant – Clink

The encapsulation plants, where the fuel is loaded into the copper canister previously described, will be an extension of the present intermediate storage facility, Clab, in Oskarshamn, meaning that the old facility, Clab, will be transformed into a new one. The stored fuel, which is kept in wet storage in pools around 50 m below ground, will be moved through a series of stations primarily above ground level, where all handling will be dry, to be dried and prepared for encapsulation, put into the canister insert; thereafter the canister will be closed by welding the lid on. Then the canister will be moved to the transport cask in which it will be transported by ship to the final repository (Figure 2).

Figure 2. Interior of the encapsulation plant Clink



Transport — Ship Sigrid

SKB has its own transport ship, Sigrid, which operates chiefly along the Swedish coastline, where all the Swedish nuclear power plants, the intermediate storage facility Clab, and Studsvik research center are situated with their own harbors. For the KBS-3 system, the ship will be used for the transport of canisters between Oskarshamn, where the encapsulation facility Clink will be situated, and Forsmark, where the geological repository will be situated. The planning is in the order of one such transport every week, of about five canisters each time.

In order to perform these shipments, a special transport cask for the canister is necessary. In this cask the canister will have acceptable conditions for the transport, such as damping, non-scratching, safety for explosions and water-tightness in case the ship would sink.

The canister will be loaded into the transport cask at the encapsulation plants and taken by truck the short distance to the harbor, where the transport cask is loaded onto the ship. The ship then sails to Forsmark, where it is unloaded at the harbour there, and taken by truck the short distance to the geological repository. The canister is taken out of the transport cask inside the geological repository.

Finding a Site — Site Investigations

In 1977 SKB began looking into the geology more generally all over Sweden with the aim to accumulate knowledge. This went on until the beginning of the 1990s. The feasibility studies were done in around ten places, until 2001. There were two municipalities, Storuman and Malå, which rejected being candidates for a repository, one through a referendum. Then SKB sent a letter to all municipalities in Sweden (around 290) asking for volunteers. Ten municipalities declared themselves volunteers for site investigations, among them Östhammar and Oskarshamn, which already had a long experience of nuclear facilities as there are nuclear power plants there.

Site Investigations at Forsmark and Laxemar

This led to the site investigations in Forsmark and Laxemar (with Hultsfred as a reserve site), which took place between 2002 and 2007. Both sites were extensively investigated with numerous techniques which had been previously tested at the Stripa mine in the 1980s, and later at the Äspö underground laboratory. Of importance was the drilling of investigation boreholes of depth from a few hundred meters to a km, which sampled the volume of interest, and in which a variety of sampling and investigations could be made. Examples include sampling of the groundwater at different depths, measurement of rock stresses, characterisation of the rock throughout the holes, and radar techniques. The drill cores were archived and characterised. Also the surface of the site was thoroughly investigated.[9]

Choosing a Site

After the completion of the site investigations in 2007, analysis and preparation for the submittal of the licence applications continued. One important milestone was to choose between the two sites. The purpose was to find the best-suited site depending on a number of parameters, such as rock properties, water situation in the rock, and others. In a way Laxemar had the advantage of being close to the encapsulation plant, thereby not requiring extensive transport by ship. But the rock investigated in Laxemar was not of the same quality as the one at the Forsmark site, where the homogenous lens and low amount of water at repository depth were deemed most important. So in 2009 SKB announced that it had chosen Forsmark as their preferred site for the geological repository.

Safeguards

The safeguard issue for a final geological repository is from a principle point of view interesting and challenging. Normally the safeguard system is based on the fact that the owner of the nuclear material in question declares it to the national and international authorities. These can then inspect the declared material at their will, at planned or non-planned inspections, or not at all. This is very different for a geological repository, as the deposited material never can be inspected again, unless the material is retrieved, which destroys the repository. Therefore, special care has to be taken in order to characterize and demonstrate that the material deposited is the right one, contains the right amount of fissile material, and has properties that make it safe for disposal (for example criticality). These parameters happen to be approximately the same as those needed for the operation and safety analysis of the facility, such as decay power, radiation field, nuclide inventory, burn-up, initial enrichment, and cooling time. SKB has a substantial research and development program on system that will determine all these parameters with acceptable uncertainties done in collaboration with several other countries, not least the U.S.

Decay Power

The decay power of the fuels is a central design parameter that determines the size of the repository and the number of canisters needed, and therefore has great economic importance. The basis for this is the temperature requirements of various parts of the system, such as the bentonite (must be below 100°C), and canister components. Work is ongoing to develop and enhance methods for this purpose.

A system to determine the decay power of the fuel to be encapsulated is being developed in broad international collaborations. Most likely, a fairly fast nuclear detection system will be used that together with the known history of the fuel and codes will determine the parameters. The method must have high throughput, as in the order of ten fuel assemblies per day must be determined, and be robust so that decisions are made unambiguously and quickly. It must also be accurate. As a compliment, a calorimeter will be used that can determine more independently and accurately the heat of the assemblies, the drawback being that it is slow, in the order of several days per fuel assembly to obtain the highest accuracy. This means it will be used for sampling purposes to "anchor" the results obtained with the faster method.

At the time of disposal, the fuel properties must be known to a sufficient level. These are:

- Decay power
- Safeguards: identify correct fuel, missing pins
- Fuel content — amount of fissile material
- Radionuclide inventory
- Burn-up, initial enrichment, cooling time
- Criticality — neutron multiplicity
- Radiation rates from gamma and neutrons
- Fission gas

SKB manages an ambitious program to develop these techniques in broad international collaborations with countries in Europe ("Spire" collaboration), Asia, and the United States. The basis is to employ both destructive and non-destructive methods, in order to achieve sufficient understanding of the fuels, develop suitable measurement methods and computer codes, and test these on real spent fuel (primarily at Clab). The ultimate aim is to develop the measurement system that will be employed at the encapsulation plants, Clink. This system must comply to the following requirements:

- permanent
- complete (all fuel assemblies)
- robust
- give unambiguous results
- complexitet in principle acceptable
- high throughput
- low uncertainty

These are, in significant ways, different from existing safeguard techniques. These are typically characterized by:

- being mobile
- sampling (non-complete)
- developed for use in the field
- low throughput

Therefore a thorough development of novel techniques is needed.

In the collaborations described above, a large number of measurements have been performed. Fifty fuel elements at Clab (SKB-50), twenty-five PWR and twenty-five BWR, have been chosen for measurements. They cover a wide range of parameters as possible, such as cooling time, burn-up, operating history etc. For example, these fifty fuel assemblies have been measured with gamma spectroscopic methods and some with calorimetry and neutron detection methods. This work is ongoing.

Fuel Chemistry

The fuel has in its normal state after irradiation the reactor very low solubility, which factors in fairly importantly in the safety analysis. In order to determine the solubility, leaking tests are done of each new fuel type that is approved in the Swedish system. Damaged fuel exposed to water due to holes in the cladding, for example, constitute a problem, which is the reason the damaged fuels are specially treated to obtain among other things a high degree of dryness.

Aging Issues

The aging of the fuels in the intermediate storage are investigated for aging changes in a program. This together with general predictions should show that the fuel assemblies can be handled in the encapsulation plants without accidents and delays.

Markings, Seals, Continuity of Knowledge

The development of marking methods and seals are done in collaboration with SSM, Euratom, and the International Atomic Energy Agency (IAEA). The canisters will be marked so that they can be identified in the whole chain from intermediate storage to the deposition hole in the geological repository. That the long-term safety of the canister is not jeopardized is crucial for the marking technique to be chosen.

The transport cask may be sealed, also with methods developed by the international organisations mentioned above.

Once the canister is welded, and the transport cask sealed, the transport process can move along with continuity of knowledge (CoK) maintained.

Licence Application Approval Process

On March 16, 2011, SKB submitted three licence applications for a permit to build a geological repository and an encapsulation plant; one for the geological repository and one for the encapsulation plant according to the nuclear act; and one for the KBS-3 system according to the environmental act.

The applications according to the nuclear act are reviewed by the Swedish regulator SSM (Swedish Radiation Safety Authority). These two applications were declared complete in early 2016, after almost five years of requests for additional information. After this declaration the formal review can begin.

The application according to the environmental act is a process through the Land and Environmental Court. This will finally result in a recommendation to the Swedish government.

The two municipalities involved, Östhammar (where Forsmark is situated) and Oskarshamn (where the encapsulation plant is planned) has the possibility of veto until very late in the process. The government could formally overrule such a veto, but this is not expected to happen in this process.

Finally the Swedish government (not the parliament) will make the decision to grant permit to construct a final repository. SKB expects this decision to be made in 2018.

This means that in effect five different bodies must say yes, and none of them no.

Remaining Issues

With a complex issue like a geological repository, where an analysis is done for at least a million years, there are naturally many issues, and some remain. The Swedish regulator SSM has requested additional work on canister issues, both mechanical calculations and issues such as the long-term corrosion behaviour of the copper. Also additional information has been requested on bentonite issues such as its behaviour in a semi-saturated state for thousands of years. Chemical erosion (disperging) in long-term scenarios where groundwater of very low salinity enters the bentonite and dissolves it, is another, as is piping, where flowing water is not absorbed by the bentonite, which can occur short-term. Generally, handling the water in the system remains fundamental.

A very important issue for design and economy of repository is the determination of the decay heat with sufficiently high accuracy, as explained above.

Continued efforts are also done on the aging properties of the fuel itself, so that it is certain that the fuel be in such a state that it can be handled by the encapsulation system at the appropriate time.

Information and Communication, Political Aspects, Opinion Polls — Public Perception

The issue of the spent nuclear fuel — or nuclear waste — has always been an important argument against nuclear power as such, in Sweden and elsewhere. As described in the introduction, this was also why the spent nuclear fuel program was started in 1970s. The transparency of the geological repository project has been of paramount importance since the outset. Virtually all reports written in the process are public, with the exception of particular facility details. The public has the possibility to visit facilities such as Äspö HRL, and more than half the population of Oskarshamn has done so. There has been a thorough ambition to inform factually. As described above, some setbacks occurred in the 1990s when two northern municipalities rejected the idea of becoming sites for investigation. On the national level there has been opposition from environmental groups; one has even proposed another method to take care of spent fuel, namely deep (order of 4-5 km) boreholes. Nationally there is support for KBS-3, although much less pronounced than in the municipalities in question now. The issue is not a very prominent one on the national level, for example in news coverage.

SKB has taken the view that whatever opinion about nuclear power one may have, the spent fuel exists and has to be taken care of. The moral argument, that is also inscribed in law, that the same generation that created and benefited from nuclear power also should solve the problem puts some constraints on the action. For example, if there is a solution that is deemed acceptable according to all criteria in law, then it would be counterproductive (indeed immoral) to look for completely different and new solutions which would take many decades to develop to the same level as KBS-3.

It is deemed important that Sweden takes care of its own spent fuel, and only its own.

Opinion polls are taken regularly at the sites Forsmark and Oskarshamn. There has consistently been a very high acceptance to have a repository in their own municipality, more than 80 percent. One of reason for this high consistency is probably the locally acquaintance with nuclear facilities, as there are already nuclear power plants here since the 1970s.

There are compensation packages agreed to, where both municipalities get access to funds for use. The one that has been chosen for the repository gets one-quarter, and the other the rest. This money is used for various good causes in the municipality, such as the extension of the internet broadband fiber grid, road construction, and industrial development, decided by a joint committee.

The regional level plays a relatively minor role in the Swedish decision system in this regard, although some environmental and planning issues are handled by the region (or county). They are not considered to have veto power in the process.

The final decision to grant permission for a geological repository will be taken by the national government (not by the parliament). There has been a lively debate on nuclear issues in Sweden at least since the 1970s, and broadly the attitude is positive towards nuclear power.

SKB takes part in numerous projects, groups and other arrangements internationally, to be able to follow the international development, inform international experts on the progress and share the best expertise. The work of SKB is reviewed by international experts regularly. One aim is to get international acceptance for the KBS-3 system, and related issues.

Finland and SKB's counterpart Posiva has adopted the Swedish KBS-3 system as theirs for the planned facility in Olkiluoto. Since the end of the 1990, there has been a detailed and formal collaboration between the two organizations. The repository designs are similar but not identical, due to differences in rock and ground water conditions etc., and some national differences. The ambition is to harmonize the requirements and design as far as possible to enhance efficiency and economy. Posiva got a construction licence for the geological repository in Olkiluoto by the Finnish government in December 2015 as a world-first.

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This paper is a summary of the presentation made at the 2016 INMM Spent Fuel Seminar.

The Current Spanish Spent Fuel Plan

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Abstract

This paper documents a review of the 2016 Spanish spent fuel plan and the strategies developed to meet it. All versions of the ENUN dual purpose casks are discussed, including a new design customized for the transportation of spent nuclear fuel in the People's Republic of China. The technical challenges faced by Ensa's Spent Fuel Cask Division in the development of the equipment required to store and transport spent nuclear fuel from Spanish nuclear power plants are discussed.

Background

During the last ten years, all Spanish public stakeholders involved in the management of radioactive waste have been working on the establishment of a national strategy that considers the particularities of the Spanish territory and its existing eight nuclear facilities, with ten nuclear reactors. The 6th Radioactive Waste Plan, developed by the government of Spain in 2004, gives priority to a centralized interim nuclear spent fuel storage facility, named ATC (Spanish acronym for Almacén Temporal Centralizado), where all spent fuel generated shall be temporarily stored, in canisters in a vault type system, until a decision on the final disposal is taken.

In order to develop the specific equipment necessary for storage, transport, and transfer of the spent fuel to the canister at the ATC, Ensa, a Spanish state-owned company with more than forty years of experience in the nuclear sector, decided to start the development of a new series of spent fuel casks specifically adapted to the challenging requirements of the recently established strategy.

For that purpose, Ensa has developed a new concept dual purpose cask named ENUN (acronym of ENsa UNiversal). Different versions of cask, following the same ENUN concept design, are currently under development by Ensa's Spent Fuel Casks Division, to meet the requirements of each Spanish nuclear power plant and its different types of spent nuclear fuel. Ensa has taken advantage of the knowhow accumulated during its more than forty years of existence in the manufacturing of large nuclear equipment, as well as previous designs of spent fuel casks developed in the 1990s for the Trillo nuclear power plant (NPP) (the Ensa-DPT) and for Japan (the HIEN-69). However, new restrictions imposed by stakeholders are technical challenges that Ensa's qualified staff are currently successfully facing. The challenges include lowering costs while maintaining the quality and safety features of previous designs, enhancing cask capabilities for the improvement of the spent fuel loading programs (like allowance of high burnup fuel) or providing solutions for the damaged fuel.

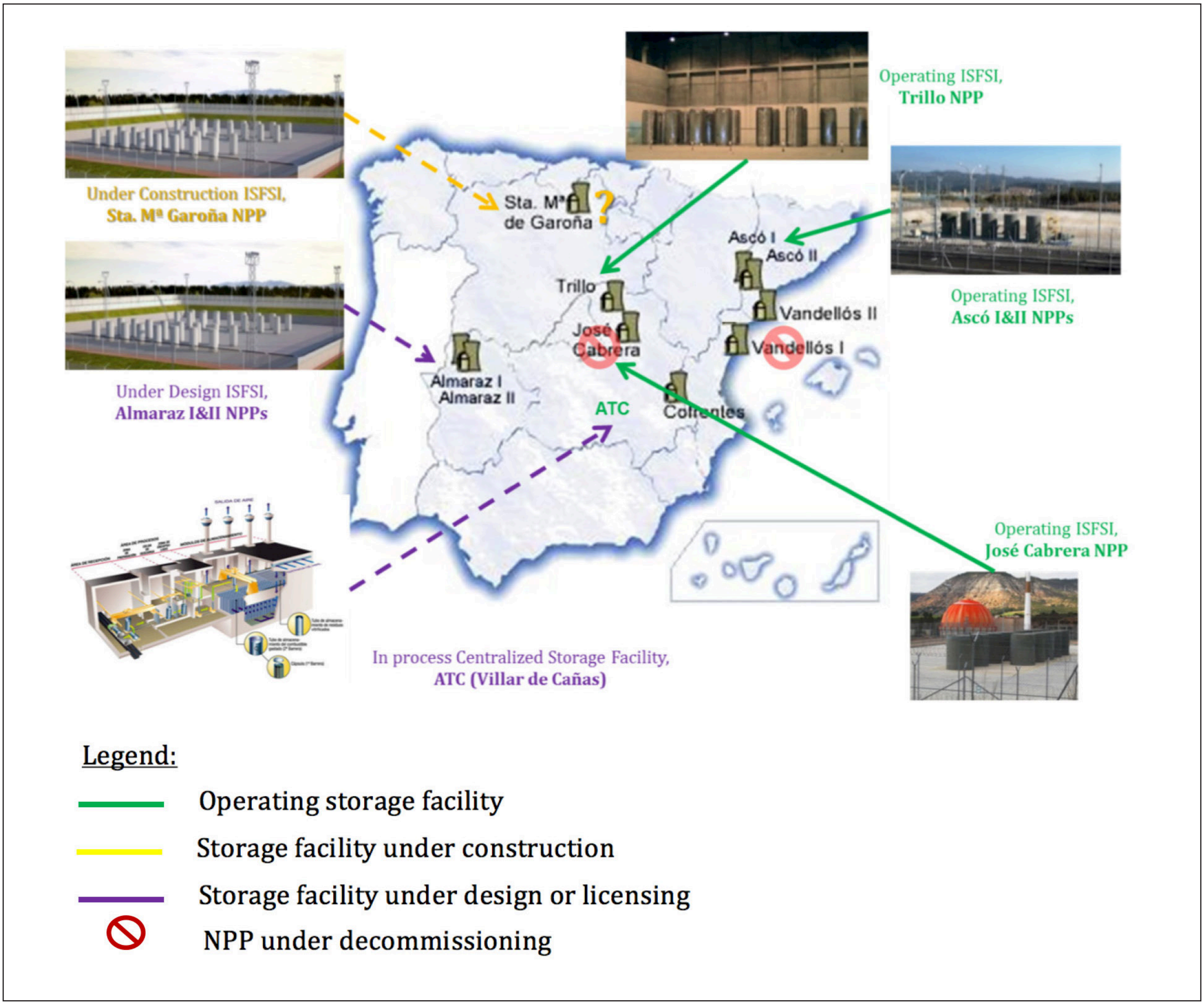
Quick Review of the Spanish Current Spent Fuel Scenario

Approximately 4,600 tons of uranium (tU) from spent nuclear fuel (SNF) are stored at the different Spanish nuclear power plants (NPPs), mostly in spent fuel pools. Three interim spent fuel storage installations (ISFSIs) are currently in operation, located at the Trillo, José Cabrera (being decommissioned) and the Ascó NPPs. A fourth ISFSI is under construction at Santa María de Garoña NPP and a fifth ISFSI is under design at Almaraz NPP. A moratorium prohibiting the construction of new nuclear power plants was dictated by Spanish government in 1984. Therefore, considering the existing NPPs, the best estimate of the total SNF that will be generated in Spain in the future, assuming each reactor operates for its forty year design life, is around 20,000 fuel elements, equivalent to 6,700 tU.

A review of the dry storage and ISFSI technology selected at these sites and the status of the spent fuel pools in the rest of the plants is presented below.

Trillo NPP is currently using the ENSA-DPT dual purpose metal cask technology, which is licensed for both storage and transport. Transportation may be done immediately after loading or after a long period of storage. The ISFSI is a concrete building with air inlets and outlets, and a capacity for at least eighty ENSA-DPT metal casks. The first ENSA-DPT cask was

Figure 1. Actual spent fuel scenario in Spain



loaded at Trillo in 2002 with twenty-eight currently loaded. The last two casks are currently being manufactured at the Ensa workshop. However, Trillo NPP operates fuel to higher burnup values than the maximum of 50,000 MWd/tU to which the ENSA-DPT was designed for storage and transport. Therefore, a new spent fuel cask design solution will be required shortly.

José Cabrera NPP and Ascó NPP (Ascó I and Ascó II) use the HI-STORM system from Holtec International. The HI-STORM is only licensed for storage, so transferring to a transportation cask (HI-STAR) will be required. The HI-STAR casks were licensed in Spain in 2009 and 2012 for José Cabrera NPP and Ascó NPP, respectively. The ISFSI design at both sites is an open concrete pad with a capacity for twelve and sixteen HI-STORM casks, respectively.

Santa María de Garoña NPP will use ENSA's ENUN 52B design technology. The ENUN 52B is a dual purpose (storage and transport) metal cask. The ISFSI being constructed at Santa María de Garoña consists of two open concrete pads, each with a capacity of sixteen ENUN 52B metal casks. The first loading campaign has been delayed to 2017, since the NPP is currently in standby because of disagreements between the Spanish government and the utility that owns Santa María de Garoña NPP.

Figure 2. Conceptual design of the Spanish centralized interim storage facility (ATC)[1]

Almaraz NPP (Almaraz I and Almaraz II) will use ENSA's ENUN 32P design technology, the ENUN 32P. The ENUN 32P is also a dual purpose (storage and transport) metal cask. An ISFSI is being designed for the Almaraz facility to store the SNF. The spent fuel pools at Unit 1 and Unit 2 will be full by 2018 and 2021, respectively. The first cask loading campaign is expected to begin in 2018.

All spent nuclear fuel in the remaining plants in Spain is stored in pools. Vandellós NPP unit II is a Westinghouse PWR design located on the northeast Mediterranean coast of Spain. The spent fuel pool will be full by 2021. Cofrentes NPP is a GE BWR design located on the east Mediterranean coast of Spain. The spent fuel pool will be full by 2019.

Finally, SNF from the Vandellós I NPP was transported to France for reprocessing after the decommissioning of the plant in 1990. An equivalent quantity of high radioactive material is expected to be returned back to Spain as vitrified waste, when the first centralized storage facility is commissioned.

Figure 1 shows the actual panorama of the SNF stored in Spain and the projected installations for dry storage to be commissioned in the upcoming years.

Spanish Spent Fuel Management Strategy

In 2004, as a result of the resolutions of the "Congressional Commission for Industry," the Spanish government was urged to create a new General Radioactive Waste Plan (6th). This new plan included, after the evaluation of different options, the start-up of a centralized interim storage facility (called in Spanish ATC or Almacén Temporal Centralizado) for the SNF and high-level waste generated in Spain and the dismantling of the NPPs that reach the end of their service lifetimes. Therefore, spent fuel management strategy in Spain considers the ATC as the centralized storage facility to receive and store all of the SNF from all the Spanish NPPs.

The ATC design is based on a vault system (see Figure 2). This concept will safely store and temporarily solve the SNF and other storage problems of high-level waste for at least sixty years, with the potential to go up to 100 years. An ISFSI attached to the main building will also be constructed to receive loaded casks from the plants and wait until conditioning the ATC canisters for transferring and storage of the SNF.

Therefore, the Spanish strategy on spent fuel management establishes that all SNF shall be shipped from the plants to the ATC in transportation casks. There is no official schedule for the first shipment of SNF to the ATC. For this reason, and based on the current inventory previously mentioned, transportation of SNF across Spain shall consider two different approaches: (a) the transport of SNF immediately after loading the fuel from the pools to the dry casks and (b) the transport of SNF after a period of storage in dry storage casks at the different ISFSIs existing at some Spanish NPPs.

When the transport casks arrive at the ATC, the SNF will wait and store them at the ISFSI attached to the main building, until they can be transferred into the canisters inside a dry hot cell. These canisters will be welded and will store the SNF during the licensed period of the ATC. These canisters shall be able to accommodate intact, undamaged and damaged fuel (inside appropriate specific cans for damaged fuel).

Restrictions Imposed by the Different Spanish Stakeholders for Storage and Transport of the SNF

As a consequence of the current spent fuel management strategy that was agreed for Spain after 2004, the following restrictions have been imposed by all stakeholders involved, for all spent fuel cask vendors that were interested to develop specific casks to remove the spent fuel from Spanish NPPs:

a) The Spanish Government: as mentioned before, after the issuance of the 6th General Radioactive Waste Plan, casks should be designed considering that spent fuel will be transported from the ISFSIs of each NPP (or directly from the pools) to the centralized storage facility (the ATC). Later on, the spent fuel will be transferred from the casks to the canisters of the vault system, for a temporary, but long, storage period. Therefore, simplicity for transport and transfer of the spent fuel is a specific requirement of this strategy.

b) Enresa (National Radioactive Waste Company): provide lower cost but safe and reliable cask solutions, able to store and transport non-damaged and damaged spent fuel.

c) Spanish Nuclear Power Plants: provide customized solutions considering the specific requirements of each of the power plants. Among those, it is interesting to highlight the following specific and challenging conditions that Ensa is currently addressing:
   - Trillo NPP: the current ENSA-DPT dual-purpose cask is designed for a maximum burnup of 50,000 MWd/tU, and actually Trillo NPP operates fuel to higher burnup values. Therefore, an enhanced cask solution capable of holding greater maximum burnup fuel is required by 2017.
   - Almaraz I and II NPPs: currently all fuel is wet stored in pools, so there is a need to free space in the pools and a cask solution for high burnup spent fuel is required by 2018.
   - Sta. María de Garoña NPP: currently all fuel is wet stored in pools, so whether the plant continues to operate or is finally shutdown, there is a need to free space in the pool or remove all spent fuel, respectively. Therefore, a customized cask solution for spent fuel is needed, taking also into account the specific dimensional and load restrictions of this plant.
   - Cofrentes NPP: currently all fuel is wet stored in pools, so there is a need to free space in the pool and for a customized cask solution for spent fuel, taking also into account the specific dimensional and load restrictions of this plant, by 2021.

Therefore, considering this scenario and all specific requirements established by all stakeholders involved in the spent fuel management in Spain, in 2006 Ensa launched an ambitious program to develop a new generic concept design for a dual-purpose cask, that later will be customized in different configurations to comply with the specific requirements of all Spanish NPPS. This project was called 'ENUN', which is the acronym of ENsa UNiversal.

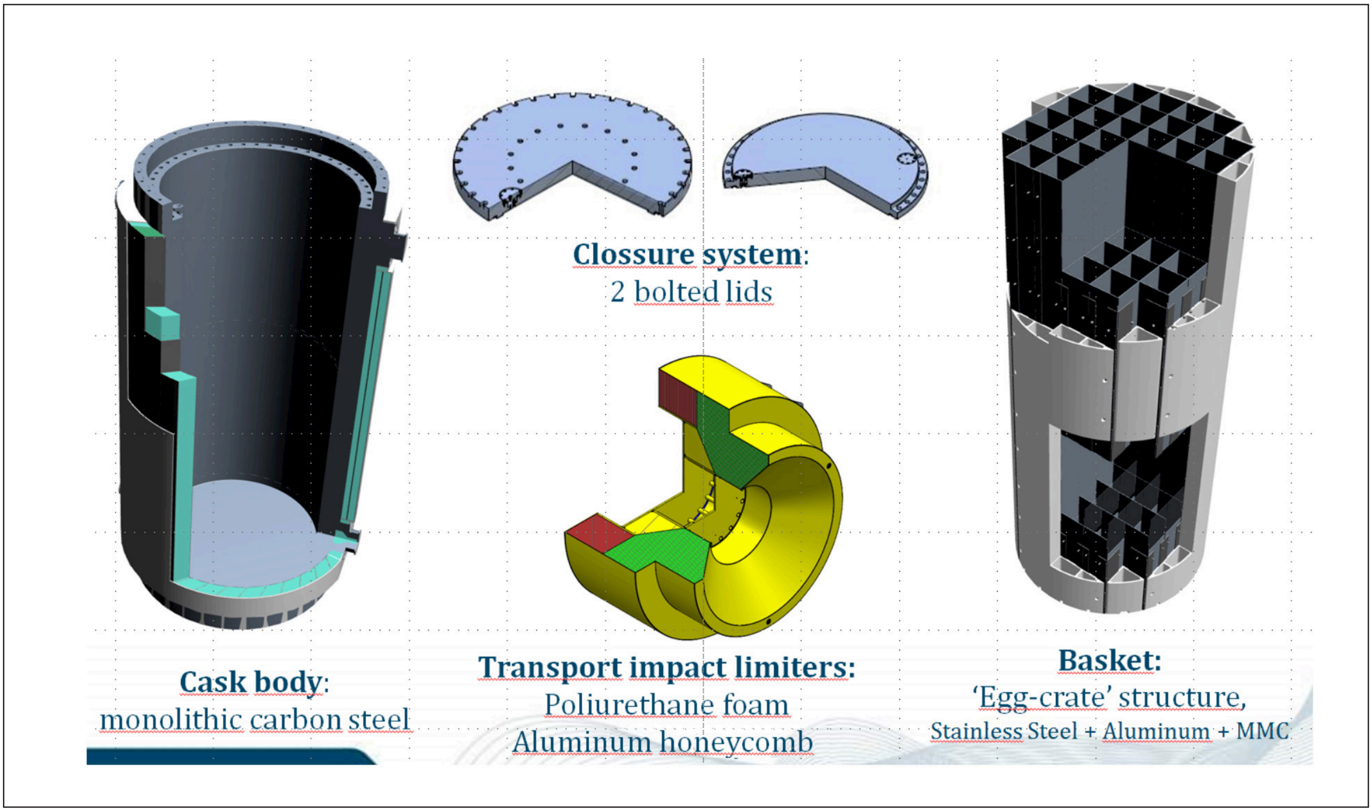
In the following pages are briefly explained the specific characteristics of all the ENUN cask developed by Ensa up to date, together with the new challenges that Ensa's Spent Fuel Casks Division will face in the short term.

The New Dual-Purpose ENUN Cask Series

The ENUN cask concept design has been conceived as a dual purpose cask for storage and transport (as a type B(U)) package, for low- and high-burnup spent fuel and canisters containing non-fuel hardware.

The cask (see Figure 3) consists of a monolithic carbon steel inner shell (made of various forging pieces) that provides the containment, gamma shielding, and structural protection of the radioactive material located inside. Surrounding the inner shell are extruded aluminium profiles that contain poured neutron absorbing material, to perform the neutron shielding safety function. To contain the system, all profiles are surrounded by a carbon steel outer shell covered by a protective coat-

Figure 3. Principal components of the ENUN casks



ing. These casks include a closure system consisting of two bolted carbon steel lids. The confinement of the inner cavity is assured by double O-rings metal gaskets that perform the sealing function of the cavity. Inside, the fuel assemblies are located in the basket cells, constituted by an egg-crate structure, assembled from stainless steel and borated aluminium (fabricated as a metal matrix composite material) plates that perform the structural, heat removal, and criticality control safety functions of the spent fuel. Finally, employed only during the off-site transportation of the cask, a pair of impact limiters with polyurethane foam and aluminium honeycomb, are bolted to the outer lid and the cask end bottom, providing the energy absorption capacity for the cask in case of any type of free drop event.

The bolted closure system has been chosen for the ENUN cask series developed for the Spanish market, because it is considered the most appropriate and versatile solution, considering the different stages the spent fuel will follow after the wet storage in the pools, as established by the Spanish spent fuel management strategy described before. In most cases, the spent fuel will be dry stored in casks in the ISFSI of each plant. After that, the cask will be off-site transported from the plant sites to the cask storage building that will be constructed contiguous to the centralized interim storage facility (ATC). After that, the spent fuel elements will be transferred from the cask to the canisters of the ATC vault system, where the spent fuel will remain stored between sixty and a maximum of 100 years, when a decision regarding its final disposal should be agreed by all stakeholders. Thanks to the bolted closure system of the ENUN dual purpose casks, the spent fuel can be safely transported and transferred through the different steps of the process, by common loading/unloading operations, and following ALARA principles in terms of radioactive doses to personnel.

Currently, three different versions of the ENUN cask concept design have been completed by Ensa. There are described below, indicated also are the fuel and the NPPs for which they are designed.

The ENUN 32P

This cask was the first ENUN developed by Ensa. It is designed to accommodate the great majority of PWR spent fuel assemblies irradiated in the Spanish nuclear power plants that are

Figure 4. Basket loading patterns of the ENUN 32P

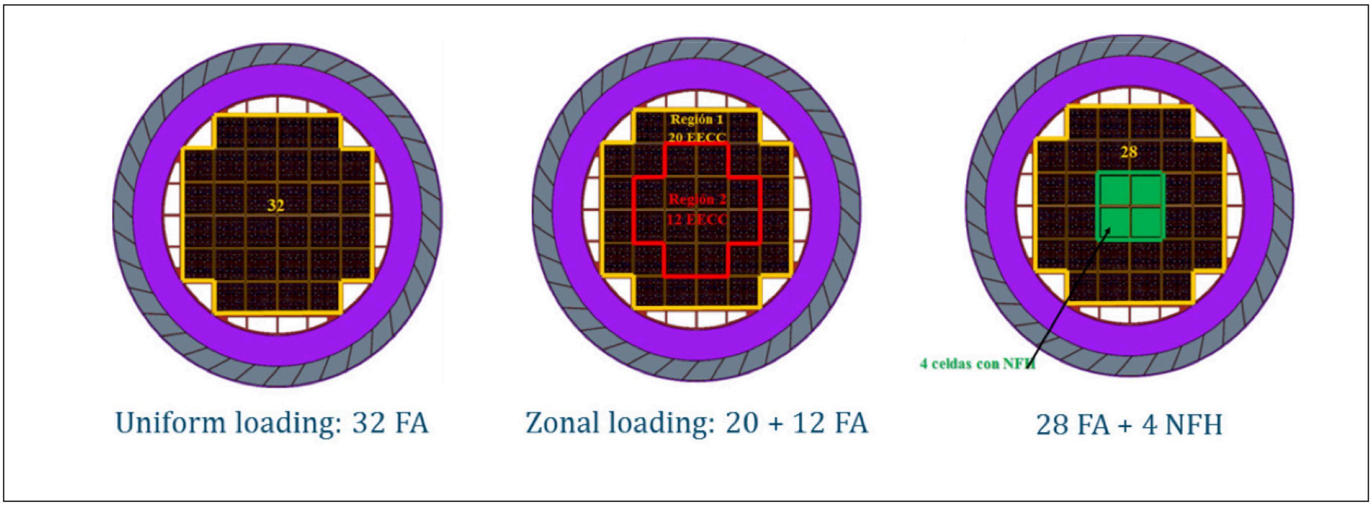


Table 1. Parameters of the authorized fuel of the ENUN 32P cask

Parameters	Value
Initial enrichment (% wt. U-235)	4.90 %
Burnup (MWd/MtU)	65,000
Minimum cooling time (years)	
a) Uniform loading	16
b) Zonal loading	
(Region 1, periphery)	21
(Region 2, center)	9
Maximum thermal load (kW)	36.2

currently in operation: Trillo, Almaraz I and II, Ascó I and II, and Vandellós II. The authorized fuel designs for this cask are the KWU 16x16 and the Westinghouse 17x17.

As indicated before, this cask includes some new features to enhance the capabilities of former ENSA-DPT developed in the 1990s exclusively for the Trillo NPP. Among those are the following:

- The basket allows two types of cell configurations, to accommodate either KWU 16x16 fuel (from Trillo NPP) or Westinghouse 17x17 fuel (from Almaraz I and II, Ascó I and II, and Vandellós II NPPs). In addition, it is also possible to accept specific canisters for non-fuel hardware like control rods and burnable poison (WABA, BPRA) components.
- The capacity has been increased from twenty-one to a maximum of thirty-two positions.
- The manufacturing cost has been reduced, mainly because of the use of carbon steel instead of stainless steel + lead, as principal materials for the gamma shielding.
- The principal parameters of the authorized fuel have been increased: the maximum design thermal load has been increased from 24 kW to 36 kW (although the cask has been designed to reject a higher thermal power) and the burnup credit methodology has been considered in the criticality assessment. These two enhanced design features allow loading spent fuel with an initial enrichment of up to 4.9 percent w.t. U-235 (instead of 4 percent w.t. U-235) and a maximum burnup of 65 GWd/tU (instead of 50 GWd/tU).

The principal bounding parameters of the authorized spent fuel for the ENUN 32P are indicated in Table 1.

The three different basket loading patterns, that allow loading all authorized spent fuel in any of the basket cells (uniform loading), load the "hottest" fuels in the central positions and the "coolest" in the periphery in order to maximize the overall thermal load of the cask (zonal loading) and the configuration for loading four canisters with non-fuel hardware in the central positions, are shown in Figure 4.

Regarding the licensing status of the ENUN 32P cask, Ensa received the Certificate of Compliance (CoC) for Storage for the vast majority of the authorized fuel irradiated in all operating PWR Spanish power plants in September 2015 from the Spanish Ministry of Industry. For the transport licensing, in February 2016 Ensa received the confirmation from the Spanish Nuclear Safety Council that the review of the Safety Analysis Report (SAR) and additionally, requested documentation, had successfully concluded, and no additional comments would be issued. Immediately after that, Ensa submitted the new official revision of the transport license documentation and the official submittal of the CoC for Transport from the Ministry of Industry was expected in late March or early April 2016. This certificate would include a limitation to transport only low burnup fuel (<45 GWd/tU). However, Ensa is currently involved in the development of a specific methodology and the performance of appropriate analysis, to justify that the cask is able to transport also high burnup fuel in safe configuration and complying with the applicable regulations as well as recommendations from the nuclear authorities and research centres. The submittal of a revised version of the SAR, including the evaluations of the high burnup fuel, is forecasted for 2017.

The licensing process of the ENUN 32P has been temporarily delayed because most of the resources from Ensa and the CSN were dedicated to a new similar project, the ENUN 52B (explained later in this paper). The 52B has higher priority because of the need to free space in the pool of Santa María de Garoña NPP.

Ensa has been recently awarded a contract by Enresa for the manufacturing and supply of ten ENUN 32P casks and corresponding ancillary equipment, for Almaraz and Trillo NNPs. Activities for material purchase and manufacturing documentation have already begun, and the first loading campaigns in both plants are forecast for early 2018.

Finally, it is very important to note that the ENUN 32P cask has been recently selected to perform a series of tests within a research program of different U.S. Department of Energy (DOE) national labs, to measure the vibrations of fuel rods under normal conditions of transport, and during different transportation modes (rail, truck, and sea). This is great news for Ensa because it indicates the confidence of international entities in the ENUN casks, to be used for in-depth research on the behaviour of the transportation of spent fuel.

The ENUN 52B

As explained before, this cask was developed in parallel to the licensing process of the ENUN 32P. It was a challenging project for Ensa because the initial schedule required that a cask be designed, licensed, manufactured and ready for loading in less than three years, due to the need for the Sta. María de Garoña NPP to free space in the pools and continue operation.

The ENUN 52B has been specifically designed to store and transport the spent fuel from BWR Sta. María de Garoña NPP. Initially, it had been licensed to load only GE-6 and GE-7 fuel designs, but it is intended to extend the scope of the design and the license to load all the fuel irradiated at the plant. This is a small cask with a capacity limited to fifty-two non-damaged fuel assemblies. The maximum load for the gantry crane of the plant is 75 tons and the distance between trunnions of the cask was limited to 2.2 m, in order to be able to lower the cask to the containment access tunnel after fuel loading.

Ensa has been able to complete the design of the cask within the tight schedule initially established. The cask has already been licensed in Spain. The CoC for storage was issued in November 2014 and the corresponding for transport in June 2015.

Ensa was awarded a contract by Enresa for the manufacturing and supply of five ENUN 52B casks and corresponding ancillary equipment, for Sta. María de Garoña NNP. The manufacturing of all the equipment is nearly completed (Figure 5). However, the casks have not been supplied yet since the NPP is currently in standby due to disagreements between the Spanish Government and the utility. The construction of the ISFSI has been delayed. The first loading campaign is expected to start in 2017.

The ENUN 24P

The last product that completes Ensa's portfolio of ENUN cask series is the ENUN 24P cask. In this case, Ensa has developed a customized solution for the Chinese spent fuel management scenario. Ensa already had previous experience in this sector, since two units of the STC-26 cask design were manufactured and supplied in 2003 for the transport of spent fuel in China. In this case, our customer needed an enhanced solution with improved design and materials that could reduce manufacturing costs as well as reduced external package dimensions to transport high burnup fuel through more than 2,000 km across China, from the Daya Bay and Ling Ao NPPs in the southeast coast to Lanzhou storage facility, in the central north of the country (see Figure 6).

This project has also presented a technical challenge to Ensa, due to the necessity to adapt the generic ENUN design to allocate AFA 2G, AFA 3G and AFA 3GAA PWR fuel designs with a maximum burnup of 57 GWd/tU and minimum cooling time of five years, reduce the outer package dimensions to only 3.3 meters and supply the cask, already licensed in Spain and in China, in less than 3 years. Since burnup credit was not allowed by the customer, the maximum capacity of the cask has been limited to twenty-four non-damaged spent fuel assemblies.

The design of the cask has already been completed by Ensa and the transport SAR was submitted to the Spanish

Figure 5. Manufacturing activities at Ensa workshop of the ENUN 52B casks

Safety Council (CSN) in May 2015, and later to the Chinese National Nuclear Safety Administration (NNSA). The customer needs the cask licensed for transport of high burnup fuel and ready for loading at the plant sites by the end of 2016, when the first loading and transport campaigns are forecasted in China.

Fabrication of only the ENUN 24P cask and its ancillary equipment has been awarded to Ensa. The manufacturing of all these components progresses adequately with great effort by all Ensa's staff to comply with the client's tight schedule.

Upcoming Challenges for ENSA's Spent Fuel Department

The spent fuel management scenario in Spain is currently demanding a lot of projects for all participants involved. As explained in this article, Ensa has made a great effort in the last ten years to develop appropriate and customized equipment for the dry storage and transportation of the spent fuel irradiated in Spanish NPPs. New projects have already been assigned to Ensa to enhance the capabilities of the ENUN cask series. Among those, the three most important projects that Ensa is facing in 2016 are:

Develop the methodology and perform specific analyses with the appropriate data from the spent fuel, to evaluate the behaviour of the PWR type ENUN casks for the safe transport of high burnup fuel (> 45000 GWd/tU). The first task within this project is to agree on the analyses methodology with the nuclear authorities and collect appropriate data from the irradiated fuel at the specific plants, in order to assurance compliance with U.S. NRC ISG-11, Rev. 3[2] and IAEA SSR-6[3] regulations, as well as taking into account recommendations issued by the U.S. NRC. When this job is completed, new revision of the SAR for the ENUN 32P and the ENUN 24P casks will be submitted for licensing, in order to extend the scope of the transportation license also to high burnup fuel.

Currently, damaged fuel is a problem for some of the Spanish power plants, as it consumes space and must be accommodated in dry storage. Ensa is currently developing a specific canister for non-leaking damaged fuel, compatible with all corresponding ENUN casks and also with the canisters from the Centralized Storage Facility (ATC), in order to simplify the process for transport and temporary storage the damaged fuel.

The design of the ENUN 52B is currently licensed for specific lots of fuel assemblies from GE-6 and GE-7 designs. These lots contain some of the oldest fuel assemblies irradiated at

Figure 6. Transportation of high-burnup fuel in China with the ENUN 24P

Santa María de Garoña NPP, which are the priority of the utility for removal from the pools. However, the design of the ENUN 52B shall be evaluated, adapted and finally licensed, to enlarge the scope of the authorized contents also to the rest of BWR fuels that have being irradiated at the plant and are currently stored in the pool. Within these designs are GE-4, GE-5, GE-8, GE-10, GE-11, GE-14, etc.

Conclusions

A quick review of the Spanish fuel management scenario has been provided, including the situation of every nuclear power plant and the interim storage facilities for spent fuel. Current Spanish strategy on the back-end of the fuel cycle demands specific solutions adapted to the requirements of each of the different plants, and to simplify the operations of the spent fuel: interim storage, off-site transportation and transfer to the canisters of the centralized storage system.

Ensa has been involved for the last ten years in the development of the ENUN cask design concept, conceived as a dual purpose cask for storage and transport. Ensa is taking advantage of its previous experience in the design of dry and wet storage equipment (casks and racks) as well as more than forty years of manufacturing of large nuclear equipment (reactor vessels, vessel heads, steam generators, etc.) as a multi-system supplier for several vendors worldwide. The result is the development of the ENUN 32P, ENUN 52B, and ENUN 24P casks, to provide customized solutions for the Spanish nuclear power plants, as well as meeting other challenging technical requirements from international customers.

Ensa is constantly improving its current designs as well as analyzing new business opportunities in Spain and worldwide, to offer enhanced technical solutions for the spent fuel management, and increasing the different types of cask in the ENUN series.

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Alejandro Palacio obtained a bachelor's degree in mechanical engineering in 2005, an MSc in mechanical engineering in 2007, and an MBA in 2015. He started his professional career in the automotive sector in 2007 as a project engineer in structural design and passive safety.

He entered the nuclear business in 2011 as a design and licensing engineer for nuclear spent fuel components. Since then, he has been deeply involved in the design and licensing activities of the new proprietary Ensa spent fuel cask designs (the ENUN series). Apart from design activities, he frequently collaborates in marketing proposals regarding spent nuclear fuel equipment and surveys the manufacturing activities of all the ENUN casks at Ensa workshop, to guarantee they strictly comply with the licensed design. Alejandro is a member of the board of the Spanish Young Nuclear Generation Network.

David Garrido obtained a mechanical engineering degree in 1997 and started his professional career in the nuclear business in 1998 as structural and thermal analyst.

A few years later, he became project manager of all the spent fuel cask fabrication at Ensa's facility. After six years of manufacturing experience, he has been leading the design and licensing team of the new Ensa's Spent Fuel Cask designs (ENUN) since 2008.

Ensa is a globally recognized multi-system supplier of NSSS components also providing a variety of services, casks and racks to plants. Ensa has a complete proven capability from design through operation, including proprietary cask designs for PWR and BWR fuel and non-fuel hardware, both for storage and transportation purposes. Ensa has manufactured and loaded all of the used fuel casks in Spain used by Enresa, both of proprietary designs and of other designs. Ensa is fully owned by the SEPI group, a holding company with major ownership over 16 public companies and more than 75.000 employees.

This paper is a summary of the presentation made at the 2016 INMM Spent Fuel Seminar.

A Snapshot of U.S. Spent Fuel Management Policy and Status

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Introduction

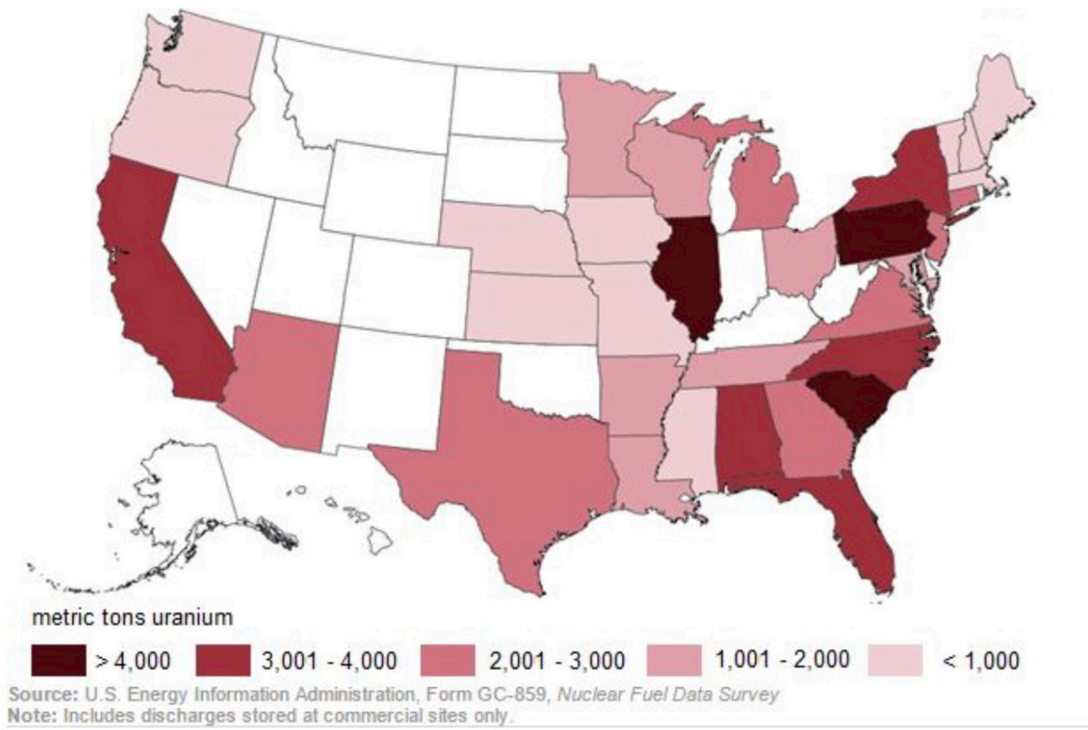
In 2009, the United States abandoned its decades-long policy of spent fuel management that envisioned all commercial and defense nuclear waste being permanently disposed of in a deep geologic repository at Yucca Mountain, Nevada, USA. Although that policy is still enshrined in law, in January 2013 the U.S. Department of Energy (DOE) announced a new integrated waste management strategy that includes a pilot consolidated interim storage facility that would store spent fuel and reactor-related greater-than Class C (GTCC) waste from so-called "stranded" locations where the reactor has been decommissioned and all that remains is the spent fuel in dry storage casks at an on-site independent spent fuel storage installation (ISFSI).[1] Following the successful operation of this pilot storage facility, the strategy calls for construction and operation of a large-scale consolidated storage facility that could accept spent fuel and GTCC waste from any commercial reactor site, and a repository that would begin operations around 2048. This strategy is consistent with the 2012 recommendations of the Blue Ribbon Commission (BRC) on America's Nuclear Future that President Obama convened after he decided Yucca Mountain was "no longer an option."[2] At the time DOE released its strategy, the agency said that legislation would be required to implement it; however, three years later, although legislation has been introduced, none has passed, and none is likely to be passed during this election year. As a result, while laws may be passed that will change spent fuel policy in the future, utilities that generate the spent fuel are currently storing it safely at the reactor sites and are planning to continue this approach to spent fuel management indefinitely.

The likelihood that spent fuel will remain in storage for a very long time has resulted in the need for revised regulations, guidance, and research and development. In 2014, the U.S. Nuclear Regulatory Commission (NRC) approved the Continued Storage Rule and its supporting Generic Environmental Impact Statement (GEIS), which contained the determination that spent fuel can be stored safely for sixty years beyond the licensed life of a reactor, and could even be safely stored indefinitely if a repository is never built.[3] This rule replaced the prior Waste Confidence findings that essentially stated spent fuel could be stored safely until a repository is operational, which was expected to be by 2025. Since spent fuel is likely to remain in storage for at least several more decades—many more decades than originally anticipated—the NRC has embarked on an extended storage and transportation (EST) regulatory program, which could include a safety and environmental analysis to support very long-term storage (up to 300 years).

Although almost all of the spent fuel currently in dry storage is "low burnup fuel," which is defined as having achieved a burnup of less than 45,000 megawatt days per metric ton of uranium (MWd/MTU), going forward, much of the spent fuel that will be placed into dry casks will be higher burnup fuel. In April 2013, DOE announced it would invest $15.8 million to study high burnup fuel that has been in dry storage for long periods of time. The Electric Power Research Institute (EPRI) leads the project team, which also includes AREVA Federal Services, AREVA TN, Dominion, AREVA Fuels, and Westinghouse Fuels.[4] Westinghouse, NAC International, the Nuclear Energy Institute (NEI), and several national laboratories are also providing support. The project is well underway.

The lack of a permanent disposal facility has also spawned interest in private companies building and operating consolidated storage facilities. Two applications will be submitted to the NRC in 2016. Waste Control Specialists (WCS), which is being acquired by *EnergySolutions,* announced plans in 2015 to build a consolidated interim storage facility (CISF) at the site of its low-level waste (LLW) disposal facility in Andrews County, Texas.[5] WCS has partnered with AREVA TN and NAC International to license, build, and operate this facility, which has broad local and state consent. The site-specific 10 CFR 72 storage license will reference AREVA TN and NAC International storage technology.

Figure 1. Commercial spent fuel in storage by state as of June 2013



Holtec International also announced plans in 2015 to build a CISF near Carlsbad, New Mexico, USA. Holtec has partnered with the Eddy-Lea Energy Alliance (ELEA) for this facility that will be built on 1,000 acres owned by ELEA near the Waste Isolation Pilot Plant (WIPP) facility.[6] Holtec's CISF will use its HI-STORM UMAX (underground, maximum capacity) storage system. Holtec also claims widespread local consent and the consent of the New Mexico governor.

Both of these companies envision that DOE would be the customer, and would take title to the spent fuel at the reactor site, transport it to the storage facility, retain title throughout the storage period, then eventually transport it to a federal permanent disposal facility. Nuclear Waste Fund money would pay for these activities, but legislation is needed to authorize this arrangement.

DOE also announced in December 2015 that it is launching a consent-based process to site federal spent fuel storage and disposal facilities.[7] This announcement followed DOE's March 2015 announcement that it would pursue a separate repository for defense waste.[8] The decision to separate commercial spent nuclear fuel and high-level waste (HLW) from defense waste is a departure from long-standing U.S. policy in which commercial and defense radioactive waste, including spent fuel, would be co-mingled in a single repository, although either approach is authorized by law. Implementing legislation is required, however, for DOE to site an interim storage facility, so until then, the department is working within its legal authority on the siting process, and making gains in developing a transportation system. The 2013 Strategy originally called for a pilot interim storage facility that would accept commercial spent fuel from decommissioned reactor sites to be in operation by 2021, a larger interim storage facility that could accept spent fuel from any site in the country to be in operation by 2025, and a permanent repository to be operational by 2048. U.S. Secretary of Energy Ernest Moniz has acknowledged that the 2021 and 2025 dates will not be met, but DOE has not formally revised its strategy with new target dates.

Current Spent Fuel Storage Status

The Energy Information Administration (EIA), the information branch of DOE, released the results of its Nuclear Fuel Data Survey in December 2015.[9] The survey revealed that as of June 2013, a total of 241,468 fuel assemblies with an initial loading weight of about 70,000 metric tons of uranium (MTU) have been discharged at 118 commercial nuclear power reactors operating in the United States from 1968 through June 2013 (Figure 1). EIA noted that "Illinois, Pennsylvania, and South Carolina have the highest amount of stored material, with more than 4,000 MTU in each state." About two-thirds of the total spent fuel in storage was discharged from pressurized water reactors (PWRs), with the balance having been discharged from boiling water reactors (BWRs). More than 99 percent of these assemblies are stored at the reactor sites, with less than 1 percent having been shipped to away-from-reactor facilities, but utilities were not required to report spent fuel assemblies that were shipped offsite.

Average burnup for all BWR fuel discharged since 1968 was 33.9 gigawatt-days per metric ton of uranium (GWd/MTU), while the average burnup for all PWR fuel discharged during that time period was 39.7 GWd/MTU. Burnup averages have increased in recent years. The average burnup for the 3,246 BWR assemblies discharged in 2013 was 44.1 GWd/MTU; for the 1,534 PWR assemblies discharged in 2013 the average burnup was 45.4 GWd/MTU. Twenty years prior, in 1993, the BWR average burnup for the 3,763 assemblies discharged was 30.3 GWd/MTU, and the average burnup for the 3,400 PWR assemblies discharged was 39.2 GWd/MTU.

EIA collected the data for the DOE's Office of Standard Contract Management, within the Office of the General Counsel, on the Form GC-859, "Nuclear Fuel Data Survey." From 1983 through 1995, EIA collected this data annually. Since 1996, EIA has collected the data three times. The last release of the data was in 2002. EIA said the information "is directly relevant to the design and operation of the equipment and facilities that the Department of Energy will use for the future acceptance, transportation, and disposal of spent nuclear fuel."

Dry Storage Status and Market Shares[10]

The dry cask storage market has seen exceptional growth since its inception in the mid-1980s. In the United States, seventy-three ISFSIs are in operation at nearly every reactor site in the United States already storing spent fuel in dry casks or having near-term plans to do so. The only exceptions are Three Mile Island (TMI) Unit 1, Shearon Harris, and Wolf Creek. One new ISFSI started operating in 2015 at Ameren UE's Callaway site. In 2016, three new ISFSIs are scheduled to start operating: V.C. Summer, Clinton, and Watts Bar. The South Texas Project is planning to begin operations at its ISFSI in 2017, and Crystal River's ISFSI could become operational in 2017.

Approximately 75 percent of the commercial spent fuel inventory is stored in pools, with the balance in dry casks at on-site ISFSIs and a small amount has been shipped offsite. According to data collected by UxC for our regular monthly publication, *StoreFUEL*, at the end of 2015, 2,277 dry storage casks had been deployed in the United States, storing 93,426 spent fuel assemblies, which is nearly twice as many casks as had been deployed at the end of 2010 (1,387 casks storing 54,046 assemblies). During 2015, 190 casks were loaded, and at least that many are expected to be deployed in 2016.

These ISFSIs are licensed by the NRC under 10 CFR Part 72, with either a general or a site-specific license. General licenses authorize the storage of spent fuel in casks previously approved by the NRC at a site already licensed by the NRC to store spent fuel at that site. Each general licensee must demonstrate that it is safe to store spent fuel at the site using the dry storage system selected. For a site-specific license, the NRC reviews the safety, environmental, security, and financial aspects of the licensee and the proposed ISFSI, and if the NRC concludes it can be operated safely, a license is issued. The license includes the specific type and quantity of material that can be stored at the site, including which system is used. Any changes to the license must be made through a license amendment or exemption. Most ISFSIs operate under a general license, with only thirteen of the seventy-three ISFSIs in the U.S. operating under a site-specific license (including one pool ISFSI, which is the GE-Morris spent fuel storage facility in Illinois). To date, seven site-specific ISFSI licenses have been renewed, most recently the ISFSI at Xcel Energy's Prairie Island Nuclear Generating Station. ISFSI licenses can now be renewed for a forty-year period. The next ISFSI license that will need to be renewed by the NRC is Dominion's North Anna ISFSI. The table below shows the status of the site-specific licenses in the U.S., sorted by initial license expiration date. Three main cask vendors are active in the U.S. dry storage industry, all of whom also have clients outside of the U.S.. The data is as of the end of 2015:

- AREVA TN is AREVA's dry storage and spent fuel transport company. AREVA TN currently markets its modular NUHOMS dry cask storage system, which is in use at twenty-seven U.S. sites storing over 29,000 assemblies in more than 800 systems. AREVA TN also has its bare fuel metal cask system in use at five sites with 175 systems deployed, bringing to a total of more than 37,700 spent fuel assemblies stored in 979 AREVA storage systems.

Table 1. Status of 10 CFR Part 72 Site-Specific ISFSI licenses – by Initial License Expiration Date
Source: UxC StoreFUEL

Licensee	Site	License	Initial License Expiration Date	Renewed License Expiration Date	Notes
Dominion	Surry	SNM-2501	2006	2046	License renewed in 2005 with exemption for an additional 20 years
Progress Energy	H.B. Robinson	SNM-2502	2006	2046	License renewed in 2005 with exemption for an additional 20 years
Duke	Oconee	SNM-2503	2010	2050	License renewed in 2009 with exemption for an additional 20 years
DOE	Ft. St. Vrain	SNM-2504	2011	2031	License renewed in 2011 for an additional 20 years
Constellation	Calvert Cliffs	SNM-2505	2012	2052	Renewed October 23, 2014 for a 40-year period
Xcel Energy	Prairie Island	SNM-2506	2013	2053	Renewed December 9, 2015 for a 40-year period
Dominion	North Anna	SNM-2507	2018	N/A	Renewal not yet submitted
Portland GE	Trojan	SNM-2509	2019	N/A	Renewal not yet submitted
SMUD	Rancho Seco	SNM-2510	2020	N/A	Renewal not yet submitted
GE	GE-Morris (wet storage)	SNM-2500	2022	2022	License renewed in 2004 for 20 years; was the first ISFSI license renewed
PG&E	Diablo Canyon	SNM-2511	2024	N/A	Renewal not yet submitted
DOE	Idaho Spent Fuel Facility	SNM-2512	2024	N/A	Renewal not yet submitted
PG&E	Humboldt Bay	SNM-2514	2025	N/A	Renewal not yet submitted

- Holtec International is best known for its HI-STORM cask technology, which is in use at twenty-nine U.S. sites, storing more than 41,000 spent fuel assemblies in more than 800 casks. Two new cask designs were deployed at U.S. sites in 2015 — the HI-STORM UMAX system at Callaway and the HI-STORM FW (flood and wind) system at Browns Ferry.
- NAC International specializes in nuclear materials transport, spent fuel storage and transport technologies, nuclear fuel cycle consulting, and fuel cycle information services. NAC systems are in use at eleven sites, storing 10,622 assemblies in 395 systems. NAC is in the process of loading more than forty casks of high-level radioactive waste (HLW) at the West Valley Demonstration Site in New York for DOE.
- A fourth cask vendor, *EnergySolutions* (ES), no longer actively markets cask systems, but ES has sixty-six cask systems storing 1,833 assemblies in use at four U.S. plants. Renewal of its VSC-24 Certificate of Compliance (CoC) has been under review by the NRC since October of 2012 and should be finalized later this year. This renewal will be the first generic cask design to be renewed by the NRC. ES also manages customized projects for utility customers, such as the Zion decommissioning project in which all of the Zion spent fuel was transferred into sixty-one of NAC's MAGNASTOR storage systems in a continuous thirteen-month loading campaign that was completed in January of 2015. In addition, Zion loaded four GTCC waste canisters. More recently, ES entered into a similar decommissioning agreement with Dairyland Power Cooperative for the LaCrosse Boiling Water Reactor.

The accompanying charts show the market shares of each vendor. Figure 2 shows the number of assemblies that are stored in casks of each vendor. Figure 3 shows the number of casks of any type that each vendor has deployed. Figure 4 shows only the dual-purpose concrete/steel storage systems that are in use. This subset is broken out separately because these types of systems are the only types being purchased today.

Over the last few years, the capacities of the storage systems have increased. A few years ago, the typical capacity for a spent fuel cask that stored PWR assemblies was twenty-four or thirty-two assemblies per cask. Now, both Holtec and NAC International have high capacity storage casks that can store thirty-seven PWR assemblies (Holtec's HI-STORM FW system), or thirty-nine PWR assemblies (NAC's MAGNASTOR system). The NRC is in the final stages of reviewing a high capacity storage system for AREVA TN as well. This system, which AREVA TN has named the NUHOMS EOS (Extended Optimized Storage), will have a cask design that can store thirty-seven PWR assemblies. Similarly, capacities for casks that

Figure 2. Assemblies in dry storage in the U.S.

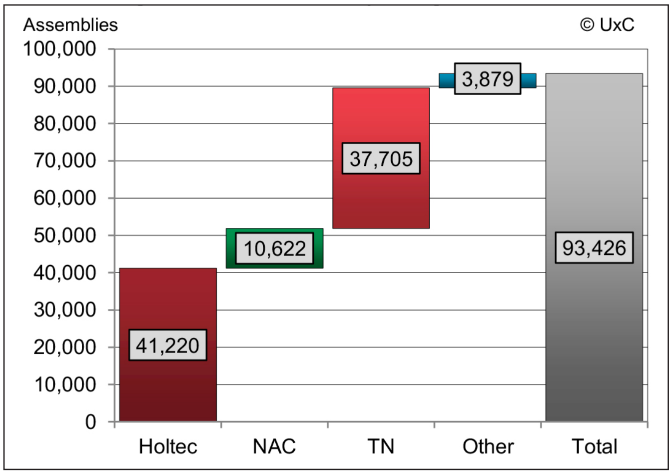


Figure 3. Dry storage systems loaded in the U.S.

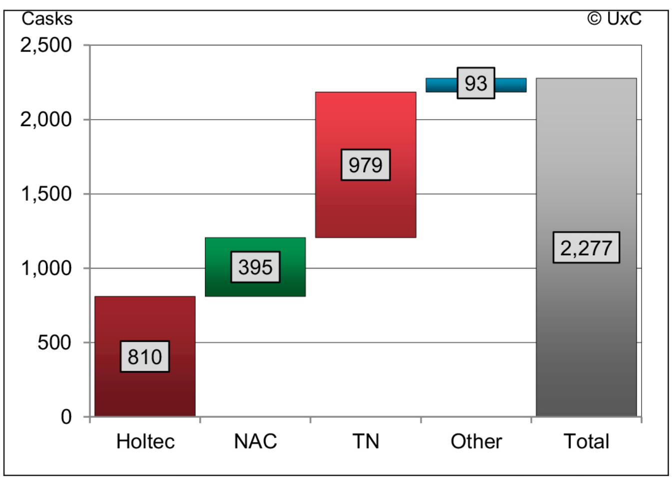
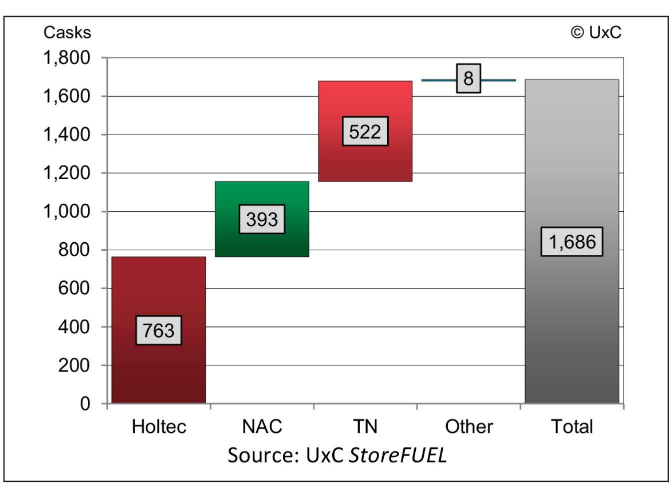


Figure 4. Dual purpose concrete systems currently deployed



Source: UxC *StoreFUEL*

store BWR fuel have also increased. A few years ago the typical capacity for a BWR cask was sixty-one or sixty-eight BWR assemblies. Now both Holtec and NAC have casks that can store eighty-nine (the HI-STORM FW) or eighty-seven (the MAGNASTOR) BWR assemblies, and AREVA TN's NUHOMS EOS system contains a high-capacity canister that can store eighty-nine BWR assemblies. The storage capacities are important to note because while AREVA TN has the most number of assemblies in dry storage, many of its systems installed are lower-capacity systems, so the actual number of casks deployed depict a different market share.

The market share in terms of number of dual-purpose concrete casks in use is depicted in the first two pie charts (Figure 5). For BWR fuel, Holtec is the market leader with almost 60 percent of the market, followed by AREVA TN with nearly 40 percent. For PWR fuel, AREVA TN is the market leader with 44 percent of the market, followed by Holtec at 26 percent and NAC International at 25 percent.

In terms of number of assemblies (see Figure 6), however, because of the variation in cask size, 64 percent of BWR spent fuel assemblies are in Holtec casks, followed by 34 percent in AREVA TN casks. For PWR fuel, largely because of the high capacity MAGNASTOR system that has been deployed at the Zion plant, 38 percent of PWR spent fuel assemblies are in NAC systems, followed by 33 percent in Holtec systems, and 29 percent in AREVA TN systems.

One final graph (Figure 7) shows the market share at shutdown sites by vendor. Here NAC International has the largest market share followed by AREVA TN and Holtec. NAC systems are in use at all of the shutdown Yankee sites, and in early 2015, the Zion plant completed loading sixty-five MAGNASTOR systems (sixty-one for spent fuel and four for GTCC waste).

Recent developments in cask technology also allow for higher burnup fuel to be placed into dry storage, and for the assemblies to be transferred into dry storage with fewer years of cooling in the spent fuel pool. Some designs only require three years of pool storage before assemblies can be transferred into dry storage.

The dry storage market in the U.S. and globally will continue to see steady growth for the next several decades. In the U.S., approximately 200 more casks are expected to be deployed in 2016 at commercial reactor sites, plus the forty-plus casks of HLW that NAC International is loading at the West Valley Demonstration Project in New York. The global dry storage market will also continue to grow as a result of decommission-

Figure 5. Dual purpose concrete systems (Vendor Shares)

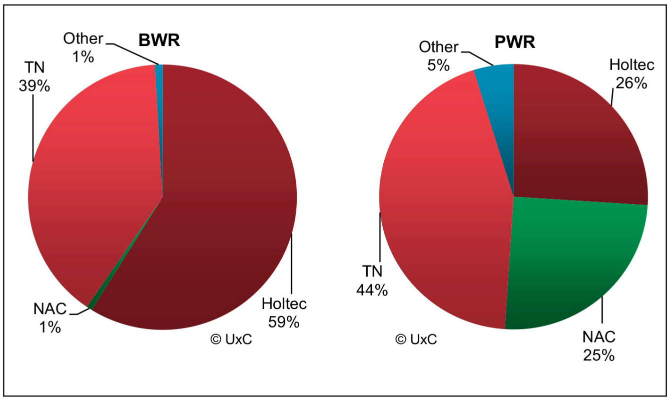
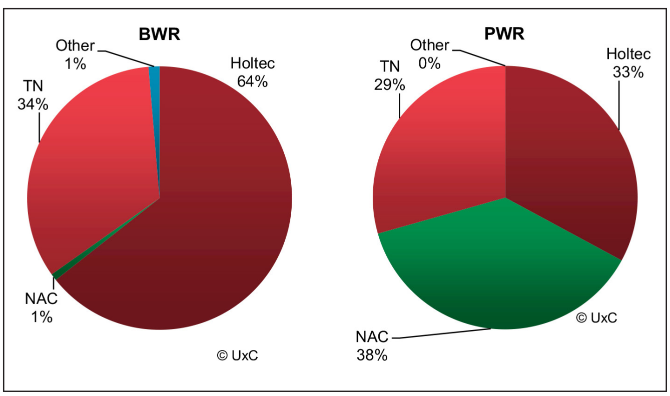


Figure 6. Assemblies in dry storage (Vendor Shares)



ing plans, particularly in Germany, delays in reprocessing plans for the countries that are pursuing it, and delays in repository programs.

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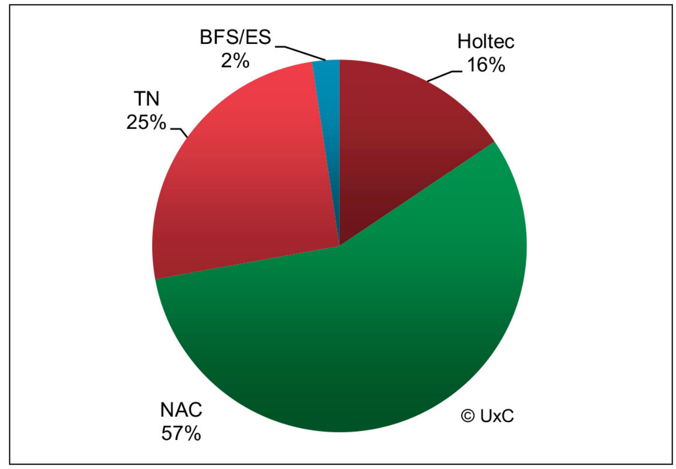
Figure 7. Market share at shutdown sites by vendor



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A New Method to Measure the $^{240}Pu_{eff}$ Mass and to Verify the Pu Fissile Content in FBR MOX Fuel Assemblies

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Abstract

This paper presents three neutron methods to measure the plutonium content in mixed-oxide (MOX) fuel assemblies that verifies the neutron multiplication and also the plutonium grade. The $^{240}Pu_{eff}$ content in MOX fuel assemblies has been measured for decades using neutron coincidence counting. The present paper uses neutron coincidence counting combined with Differential Die-Away Self-Interrogation (DDSI),[1,2] and multiplicity counting. The paper will focus on fast breeder reactor (FBR) MOX assemblies; however, the method also applies to light water reactor (LWR) MOX fuel assemblies. The plutonium recovered from nuclear weapons dismantlement is being converted into MOX powder for fuel assembly fabrication, and it is important to establish that the plutonium in the fuel assembly is from weapons grade plutonium (WPu) versus reactor grade plutonium (RPu). A key isotopic indicator between the two types of plutonium is the $^{240}Pu_{eff}$ fraction that is typically 4-7 percent in the weapons grade material and 15-40 percent in the reactor grade material. This paper presents a ^{3}He tube-based detector design that is optimized for determining the $^{240}Pu_{eff}$ fraction in the plutonium. The primary purpose of this paper is to evaluate three analysis enhancements to standard neutron coincidence counting for the verification of the mass of WPu in FBR fuel assemblies.

Introduction

The accurate measurement of the plutonium in fresh mixed oxide (MOX) fuel assemblies is of key importance for the safeguarding and accountability for the plutonium fuel cycle and for the dismantlement of nuclear weapons. The plutonium from reactor fuel reprocessing plants and plutonium weapons dismantlement activity is typically recycled into MOX powder, fuel rods, and assemblies for harvesting the energy potential in both light water reactors (LWR) and fast breeder reactors (FBR). The primary purpose of this paper is to evaluate three analysis enhancements to standard neutron coincidence counting for the verification of the mass of WPu in FBR fuel assemblies.

The mass of the plutonium in the rods and the un-irradiated assemblies has been measured for several decades using neutron coincidence instrumentation.[3,4] The spontaneous fission rate (SF) from the ^{240}Pu is the dominant source of coincidence neutrons from fresh fuel, and the additional SF neutrons from the other plutonium isotopes is described by Equation 1.[5]

$$^{240}Pu_{eff} = 2.52\ ^{238}Pu + {}^{240}Pu + 1.68\ ^{242}Pu. \qquad (1)$$

The ratio of the different plutonium isotopes is typically obtained from destructive analysis (DA) or high resolution gamma-ray spectrometry (HRGS).

Background

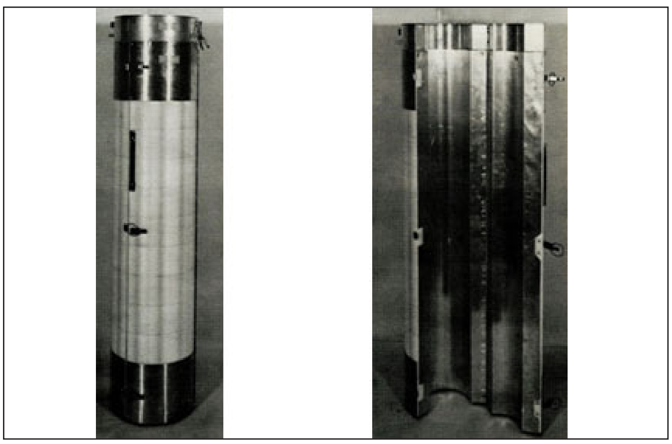
Examples of past instrumentation to measure the plutonium in MOX fuel pins are shown in Figure 1 for the Fast Flux Test Facility (FFTF) FBR fuel pins.[3] The ^{252}Cf active neutron-based rod scanner on the left provided the plutonium isotopic ratios and the pellet-to-pellet uniformity via gamma-ray spectrometry,

Figure 1. Two NDA systems used at the Hanford, FFTF fabrication plant in 1972 to measure the plutonium mass and isotopic ratios for the MOX fuel rods. The left system is a ^{252}Cf active/passive gamma scanner for pellet measurements, and the right system is a neutron coincidence system for the $^{240}Pu_{eff}$ mass.[3]

Table 1. Plutonium mass loadings and isotopic ratios for the FFTF FBR Reactor[6]

Assembly ID	MOX [g]	Pu [g]	^{238}Pu wt [percent]	^{239}Pu wt [percent]	^{240}Pu wt [percent]	^{241}Pu wt [percent]	^{242}Pu wt [percent]
16483	37428	7396	0.06	86.35	11.82	1.56	0.21
16471	37436	8386	0.05	86.85	11.62	1.27	0.21
8195	37669	9108	0.06	86.47	11.67	1.62	0.19
8255	37589	9734	0.06	85.78	11.78	1.15	0.23

Figure 2. The UFBC neutron coincidence counter used to measure FBR assemblies at the FFTF reactor site in 1984[6]



and the ^{3}He tube neutron coincidence counter (right) provided the $^{240}Pu_{eff}$ mass. The more accurate isotopic ratios from destructive analysis were used to calculate the total plutonium mass based on the $^{240}Pu_{eff}$ measurement. The system shown in Figure 1 was installed by LANL at the FFTF fabrication plant at Hanford, WA in 1972.[3] The precision for the coincidence measurement was ~ +/- 0.6 percent in 1000s for the individual pins.

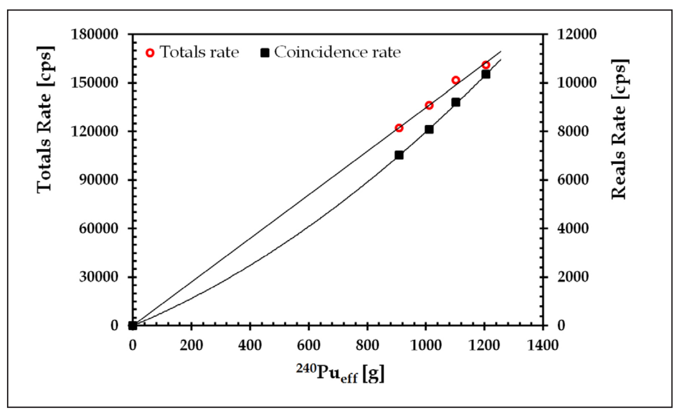
The measurement activity at the FFTF reactor site was extended to full FBR fuel assemblies in 1984 for the purpose of calibrating the Universal Fast Breeder Counter (UFBC) shown in Figure 2.[6] This detector was used by the IAEA for coincidence neutron counting to measure the $^{240}Pu_{eff}$ mass in the assemblies where Table 1 provides the plutonium mass and isotopic ratios for the assemblies. Note that the ^{240}Pu fraction was ~11.8 percent of the plutonium that is an intermediate blend between low burn-up and reactor grade. Natural uranium was approximately 75 percent of the heavy metal mass in the pins.

The UFBC detector contained 12 ^{3}He tubes (4 atm) that were inside Cd sleeves that surrounded a 15-mm thick polyethylene annulus to provide a die-away time of 21.6 μs and an efficiency of 7.0 percent for a ^{252}Cf source in the center of the sample cavity.

Table 2. FFTF subassembly plutonium mass values and counting rates using the UFBC[6]

Assembly ID	Pu [g]	$^{240}Pu_{eff}$, [g]	Totals Rate [cps]	Reals Rate [cps]	R sigma [cps]
16483	7395.9	908.23	122118	7025.7	37.4
16471	8385.7	1011.6	136053	8069.4	47.6
8195	9108.4	1102.7	151542	9192.6	52.9
8255	9733.7	1205.6	161128	10348.1	68.9

Figure 3. The UFBC totals and Reals (coincidence) rates for fresh FBR fuel assemblies at the FFTF reactor site (1984)[6]



The counting rates for the FFTF subassemblies are given in Table 2 where the coincidence rate (Reals) corresponds to a 64 μs time gate. The totals counting rate of ~150,000 cps had a negligible statistical uncertainty for a 600s count.

Figure 3 shows a plot of the Totals and Reals (coincidence) rate as a function of $^{240}Pu_{eff}$ mass, and we see that the Totals rate has a linear relationship to the $^{240}Pu_{eff}$ mass; whereas, the Reals rate is increasing with mass at a faster rate. The isotopic mixture is approximately the same for all of the subassemblies and this results in the linear behavior of the Totals rate versus the plutonium mass. The Reals rate shows the nonlinear increase with plutonium mass because of leakage multiplication (M) in the assemblies in which enhances the Reals rate more strongly than the Totals.

Table 3. Cd ratio neutron coincidence measurements of FFTF fuel assemblies using the UFBC *All rates are in cps

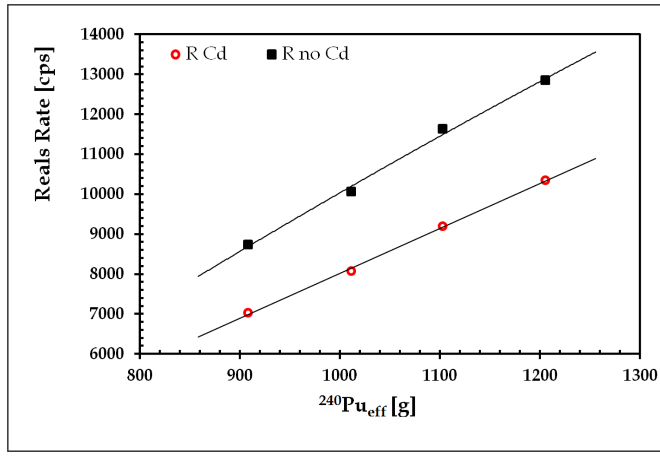
Assembly ID	T Cd*	R Cd	T no Cd	R no Cd	R/T Cd	R/T no Cd
16483	122118	7027	131154	8734	0.0575	0.0666
16471	136053	8069	145981	10055	0.0593	0.0689
8195	151542	9193	163182	11636	0.0607	0.0713
8255	161128	10348	173472	12851	0.0642	0.0741

The measurements of the FBR assemblies were performed both with and without a Cd liner on the wall of the sample cavity to provide a change in thermal-neutron reflection back into the fuel assemblies. The thermal neutrons caused fission reactions in the plutonium fissile isotopes in the exterior fuel pins. However, the induced fast neutrons from the fission reactions penetrated the entire assembly. Table 3 presents data for the same assemblies where the detector was modified by removing the Cd liner next to the fuel assembly. The Cd ratio for the Totals rate was ~1.074 corresponding to the increase in the multiplication, and the increase for the Reals rate was ~1.25 because of the boost in the coincidence probability for the multiplied reactions. We see that the R/T ratio increases as the plutonium mass increases. Note that the efficiency of the detector did not change with the Cd removal because of the second annulus of Cd surrounding each ^{3}He tube. More recent applications of the Cd ratio method have been applied to fuel rod clusters.[7]

Figure 4 shows the increased Reals rates versus the $^{240}Pu_{eff}$ mass as a result of the thermal-neutron induced fission reactions for the no Cd measurements. The R/T ratio changed by 15 percent in going from the Cd measurement to the no Cd measurement, and the R/T ratio changed by ~12 percent in the transition from the low plutonium mass assembly to the high mass assembly. The apparent linear fit of the data is only valid for the limited mass range shown in the figure and the trendlines do not extend to the zero mass origin. The Reals gate was set at 64 μs for both curves, so the no Cd data has a smaller gate fraction than the Cd data.

Typical FBR fuel assemblies will have fuel isotopic ratios that are approximately the same for all of the subassemblies in a given reactor, but there will be several different total Pu masses in the assemblies so that the reactor core can be loaded to obtain a flat power profile. However, the plutonium isotopic ratios can change from weapons grade to reactor grade

Figure 4. The UFBC Reals rates for the FFTF fuel assemblies with and without a Cd liner



and the reactor will still operate. Thus, the present paper will evaluate neutron NDA techniques to verify the plutonium grade in the fuel assemblies

Known M Verification

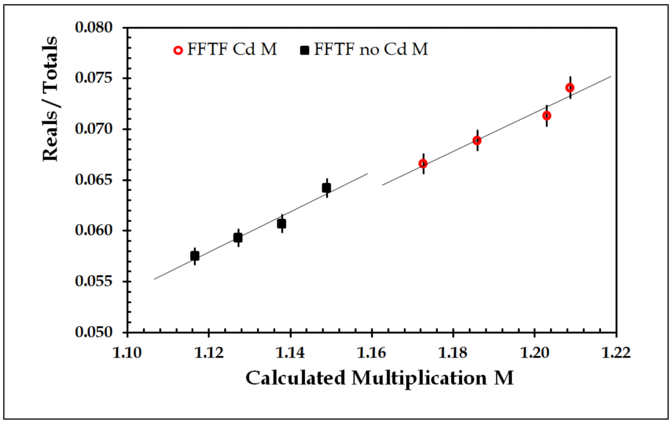
The FBR fuel assemblies have a declared, known, and fixed geometry, pin configuration, and mass. Thus, the change in multiplication becomes a function of the isotopic content where the Pu dominates the leakage multiplication (M). The M has two primary components as a function of time following the trigger neutron reaction.

1. Fast-neutron multiplication where the source neutrons have a fission energy distribution prior to collisions with hydrogenous moderator. These induced fission reactions have a sub-microsecond time delay from the trigger reaction.
2. Thermal-neutron induced fissions where the neutrons have migrated into the HDPE moderator and returned to induce fission reactions in the fuel assembly. These slow neutrons will be in the time region extending out to ~100 μs.[8]

The measurement of the time distribution of the neutrons will include the slowing down time interval from the ^{3}He detector neutron moderator. We have designed a detector with a fast die-away time to reduce the detector moderating time and to reduce the statistical error by virtue of the smaller coincidence gate setting.

For the case of the FBR fuel assemblies, we can determine the $^{240}Pu_{eff}$ mass via the standard "known alpha" analysis in INCC[9] using the declared isotopic ratios. Both the Singles (S)

Figure 5. The UFBC Reals/Totals ratios for the FFTF fuel assemblies with and without a Cd liner. The error bars reflect the statistical and estimated systematic uncertainty totaling ~1.5 percent.

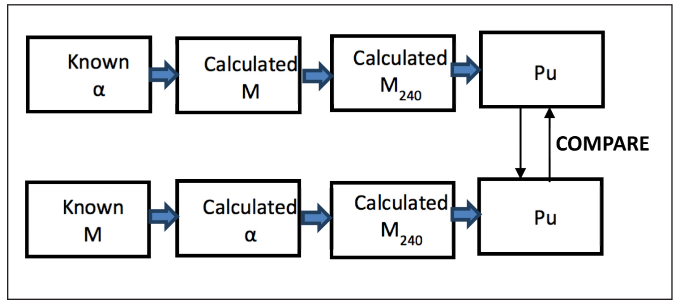


and Doubles (D) rates are measured and used to solve for M. Note that we have now changed the names corresponding to the historical data from Totals to Singles and Reals to Doubles to conform with current multiplicity nomenclature.

The R/T (i.e. D/S) ratio is approximately proportional to M for the FBR fuel assemblies (see section 7). Figure 5 shows the measured R/T ratios compared with the calculated M via the known alpha method for the FFTF assemblies. We see that the value of M increases as the Pu mass increases and that the no Cd case has higher values for M than the Cd case. The change in the R/T ratio as a function of the Cd liner change will become one of the diagnostic tools when evaluating diversion sensitivity.

For the same measured data set, we can solve for the $^{240}Pu_{eff}$ mass using the "known M" calibration where the known M calibration is determined from the measurement of "standard" assemblies or from MCNPX calculations. We then use the declared Pu isotopic ratios to calculate the $^{240}Pu_{eff}$, and compare the $^{240}Pu_{eff}$ results from the two calculation paths to verify the consistency of the isotopic mixture in the fuel assembly as illustrated in Figure 6. Both of these analysis paths are contained in the INCC software code. If the $^{240}Pu_{eff}/^{239}Pu$ fraction is falsified, the results will not agree. The sensitivity of the change in M to potential variations in the number of rods and the isotopic content in the plutonium is needed to quantify this verification tool. However, such activity is beyond the conceptual scope of this paper, but has been proposed for future research.

Figure 6. Calculation paths for determining the total Pu based on the "known alpha" path compared with the "known M" path to verify the consistency of the declared Pu isotopic ratios.



Diversion Scenarios

Three NDA evaluation techniques have been considered for the verification of the plutonium grade in FBR MOX fuel assemblies. The standard coincidence counting technique measures the S and D rates to determine the $^{240}Pu_{eff}$ in the sample. The isotopic content is input from declarations or HRGS. We will focus on the diversion cases where the isotopic ratios are misdeclared for fuel pins in the interior of the assembly that are not accessible to HRGS verification because of self-attenuation. To circumvent this problem, Lebrun and Berlizov[10] have proposed a passive gamma-ray collimated detector for verifying the absence of reactor grade fuel pins in MOX assemblies using HRGS signatures. The present paper will only cover neutron coincidence/multiplicity techniques.

The evaluation of the neutron measurement enhancements include:

1. The use of both the "known alpha" and the "known M" calibrations for the FBR assemblies (both with and without Cd)
2. The use of the Rossi-alpha analyses (DDSI) to evaluate the multiplication process
3. The use of the multiplicity Singles, Doubles, and Triples rates in the analysis to extract α, $^{240}Pu_{eff}$ mass, and M)

The diversion scenarios that will be considered include:

1. The simple removal of WPu pins and the substitution of depleted uranium pins
2. The removal of WPu pins and the substitution of RPu pins
3. The removal of WPu pins and the substitution of RPu and/or dummy pins that have added neutron emitting isotopes such as ^{241}Am and ^{244}Cm to give S and D rates that simulate the correct multiplication and WPu mass

In all cases, we assume that the exterior pins in the assembly are WPu because HRGS measurements could identify the substitution on the exterior. However, the verification of the WPu content also works for diversion on the perimeter of the assembly.

4.1 Standard Coincidence Measurements

Standard neutron coincidence counting identifies the case 1 diversion path because both the S and D rates drop below the required calibration rates by the removal of WPu. The neutrons penetrate the entire assembly so there are no hidden regions. The statistical precisions are much better than 1 percent in a 10-minute measurement using standard neutron counters such as the Universal Fast Breeder Counter (UFBC).[6] The measured rates in Table 3 for the UFBC have an error of 0.01 percent for S and 0.58 percent for D for the FFTF assemblies with a 10-minute measurement. The room background singles neutron rates were measured and found to be less than 1 percent of the singles rates from the MOX fuel assemblies.

4.2 Known M Analysis Application

If the WPu rod diversion is replaced by a smaller number of RPu pins to restore the correct D rate, the S rate will be different than what is required to get the correct multiplication. Additional radioactive sources would need to be added to the pins to get the S rate to correspond to the D rate. The $^{240}Pu_{eff}$ mass per pin from RPu is about five to six times higher than for WPu. A direct RPu pin substitution changes the D/S ratio, and the two analyses methods illustrated in Figure 6 do not agree. However, it has been pointed out in the past that the known alpha and known M result could be artificially made to agree by introducing into the pins sources of singles neutrons such as ^{241}Am and doubles neutrons such as ^{244}Cm so that the D/S ratio gives the correct M. However, this rather complex diversion scenario is prevented by the Cd ratio measurement described above, because the multiplication is changed by the removal of the Cd. Thus, the D/S ratio would require a different mixture of the diversion neutron sources for the Cd and no Cd cases. The capability to change the detector efficiency by a removable Cd liner blocks the diversion path of artificially customizing the D/S ratio to provide the correct apparent M.

For this method, we assume that the Cd ratio measurement would be made routinely for every assembly by having a removable Cd liner or by having short tandem detectors where the top detector (~20cm long) always had a Cd liner and the bottom detector was always bare. Note that the FBR fuel active length is ~100 cm long.

Whenever the S rate is used for a measurement, the control and measurement of the singles background is required. However, for an FBR type fuel assembly, the singles rate is very high (~150,000 cps for FFTF measurements[6]), and the background of less than 1,000 cps was measured prior to each assembly measurement. Note that the location of the target fuel assembly is a critical issue during the background measurement.

The singles rate is also a function of the alpha value and in the "known alpha" evaluation in INCC, and the isotopic ratios are used to calculate the alpha component for the singles and doubles. The sensitivity of the known M method to variations in the plutonium isotopic ratios and the alpha value is the subject of future proposed work.

4.3 Concept for the Verification of the Plutonium Grade Using the Neutron Die-away Distribution

To further investigate the resistance of the neutron coincidence measurements to the substitution of RPu for WPu, we have evaluated an advanced method[1,8] to verify the FBR assemblies. The method is to measure ratio of the fertile plutonium content in form of $^{240}Pu_{eff}$ and the fissile plutonium content using the passive neutron coincidence measurement combined with the DDSI (differential die-away time self-interrogation) analysis. The DDSI method measures the Rossi-alpha distribution (RAD) corresponding to the neutron time distribution following each measured neutron pulse. Figure 7 illustrates the conceptual DDSI distribution for the case where the sample is surrounded by a polyethylene moderator. We have designed the present detector to have a removable Cd liner on the inside wall of the detector cavity to change the thermal-neutron reflection back into the sample. The early time period of the RAD is dominated by fast correlation events that occur between two neutrons from the same fission event or any two neutrons from a fast-neutron induced fission chain prior to moderation in the HDPE. For FBR assemblies, events from SF and fast-neutron IF contribute more than 75 percent of the Doubles correlations registered in this early time domain. Whereas, the late time region is composed of slow correlation events comprised of two neutrons from subsequent fission events of which at least one is caused by induced fission reactions from the thermal-neutrons that are reflected back into the sample. To a lesser

Figure 7. The neutron time distribution (RAD) following each detected neutron where the counts in the short gate are dominated by spontaneous fission (>75 percent) and the long gate by thermal-neutron induced fission8

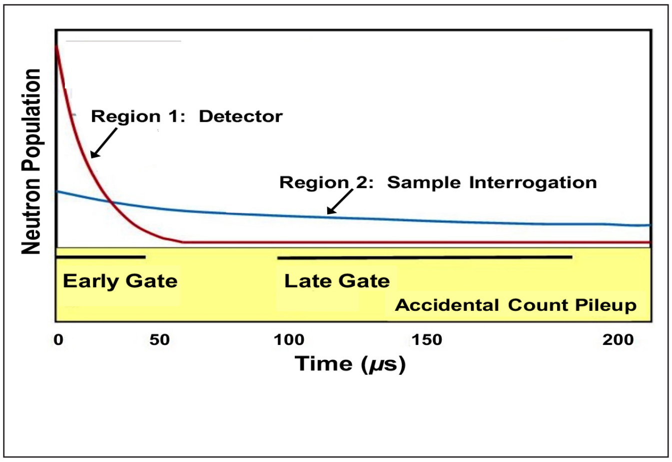
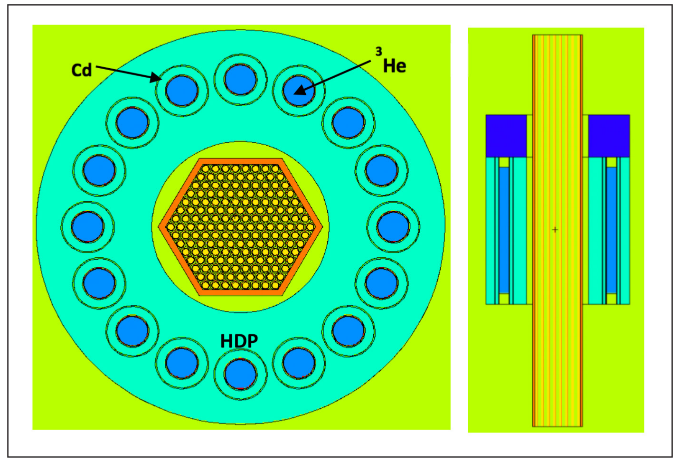


Figure 8. Cross-sectional top view (left) and cross-sectional side view (right) of the MCNPX design for evaluation of the DDSI method as applied to FBR fuel subassemblies



extent, such slow correlations also contribute to the early time domain of the RAD, but their fraction (depending on the detector design) typically does not exceed 15-25 percent. The late time extension shown in the RAD is the result of two neutron slowing-down intervals; one in the detector and one from the time delay caused by the scattering of the thermal neutrons in the HDPE surrounding the assembly. The double thermal-neutron periods extends the RAD to the later time regions.

The ^{240}Pu is the primary source of the SF neutrons, and these fast neutrons tend to populate the early time region in the RAD. Whereas, for the late time regions, the thermal-neutron induced fission reactions are reduced by the absorption of neutrons in the ^{240}Pu. The change in the fast to slow RAD ratio is about 18 percent for RPu versus WPu with uniform distributions. Our subsequent MCNPX calculations have shown that this signature to identify the WPu fraction almost disappears when the perimeter rows of pins are all WPu because the thermal neutrons lack penetrability and do not reach the inner pins — an Achilles heel for diversion detection! The enhanced fast neutrons from the perimeter induced fissions penetrate to the interior, but the neutron energies are high enough to induce fission reactions in the RPu as well as the WPu at approximately the same rate.

5. MCNPX Detector Design for FBR Assemblies

The optimum detector for measuring the $^{240}Pu_{eff}$ coincidence rate as well as the early and late gate ratios for the fissile components should have a short die-away time as well as a reasonably good efficiency for coincidence counting. Our current design focused on using ^{3}He tube detectors because of their good efficiency, stability, and proven ability to operate in the high gamma exposure from FBR subassemblies.

Figure 8 shows the detector design configuration where the focus was on having a short die-away time so that the fast-neutron events are well separated from the thermal-neutron return counts. There are sixteen ^{3}He detector tubes surrounding the FBR subassembly. Each tube is surrounded by an annulus of HDPE that is 7 mm thick, and the HDPE annulus is surrounded by a Cd sleeve to limit the migration range of thermal neutrons that are counted. The key parameter in the design is the thickness of the HDPE annulus that will dominate both the die-away time and the efficiency. The annulus thickness of 7 mm provided a die-away time of 6.2 μs and an efficiency of 4.6 percent. The ^{3}He tubes are at a 10 atm pressure to decrease the die-away time for a smaller accidental rate and to increase the counting efficiency.

For the MCNPX analysis, the Japan Monju FBR subassembly geometry was selected as being a typical FBR assembly with respect to geometry and pin configuration, but the approach could apply to any FBR assembly. The plutonium isotopic ratios were selected to cover the fuel burn-up range of interest that extends from weapons grade to reactor grade. By keeping the detector height short (30 cm active length), we have reduced the effective mass of the fuel assembly that is in the counting zone, and that reduces the counting rate and makes the system more portable.

5.1 MCNPX Result – Detector Design

The primary objective of our calculations was to simulate a realistic assay of FBR type subassemblies, and how the detector response varies with changing plutonium grade, i.e., the isotopic composition. The initial MCNPX v. 2.7c[11] calculations were used to simulate one-hour long measurement of five Monju type FBR subassemblies, all with identical geometry and plutonium mass but differing isotopic composition.

The schematic cross-sectional view of the simulated detector and fuel assembly is depicted in Figure 8. It comprises 169 individual fuel pins organized in a hexagonal matrix and encapsulated in a 5-mm thick stainless steel (SUS 316) enclosure. The MOX core of the fuel pins has a diameter of 5.4 mm and height of 93 cm and is encapsulated in a 0.55-mm thick stainless steel cladding (also SUS 316). The distance between centers of two neighboring pins is 8 mm. The ~1-mm gap in between the fuel pins is considered to be filled with air. For actual FBR assemblies this small air gap is maintained by an SUS wire wrapped in a heliacal shape that is welded to the fuel rods.

The simulated DDSI detector consists of HDPE annulus that houses sixteen individual ^{3}He tubes (placed in their own 7-mm thick HDPE annulus and wrapped in 1-mm thick Cd liner). Its inner radius is 7.12 cm, outer radius 16.42 cm and height of 35 cm. In the horizontal plane, the centers of ^{3}He detectors are positioned 12.27 cm from the center of detector (i.e., fuel assembly) with the angular spacing of 22.5°. The minimal thickness of HDPE between the sample cavity surface of the DDSI detector and the Cd liner of each ^{3}He detector is 3 cm.

The isotopic composition of individual simulated FBR fuel assemblies can be found in Table 4. The total masses of plutonium and ^{238}U were held constant (6024 g) for the initial calculations. For the assemblies, realistic Pu isotopic compositions were considered, including fractions of ^{238}Pu, ^{241}Pu, ^{242}Pu and ^{241}Am. The assembly group contain 5 cases corresponding to different levels of blending of weapons grade and reactor grade plutonium, with fuel assemblies in the range from 100W (100 percent WPu) to 100R (100 percent RPu).

To simulate the RAD as measured by the detector, we used the MCNPX code as the main simulation framework with an array of F8 capture tallies that count real coincidences in gates with predefined width and pre-delay. Coincidence gates for this work were each set to be only 2-μs long and by means of pre-delay always offset from each other by 2 μs. In this way, an interval of interest (0-120 μs in the Rossi-alpha space) has been fully covered by sixty consecutive coincidence gates. Dependence of the doubles rate in each such gate as a function of the corresponding pre-delay is then identical to the real part of the RAD with 2-μs binning. Accidentals rate can then be determined from the singles rate using standard relationship where A = *gate width* x (*singles*)^2.

Table 4. Isotopic composition of Monju-type FBR fuel assemblies simulated in this work

Isotope	100W	75W	50W	75R	100R
^{238}Pu [percent]	0.00014	0.0988	0.198	0.296	0.395
^{239}Pu [percent]	16.67	14.76	12.85	10.94	9.02
^{240}Pu [percent]	1.07	1.92	2.78	3.63	4.48
^{241}Pu [percent]	0.037	0.52	1	1.49	1.97
^{242}Pu [percent]	-	0.31	0.619	0.929	1.24
^{241}Am [percent]	0.025	0.189	0.353	0.517	0.681
^{235}U [percent]	0.14	0.14	0.14	0.14	0.14
^{238}U [percent]	70.21	70.21	70.21	70.21	70.21
^{16}O [percent]	11.86	11.86	11.86	11.86	11.86
$^{239}Pu_{eff}$ [g]	5647.8	5266.7	4878.4	4482.8	4079.6
$^{240}Pu_{eff}$ [g]	362.4	915.7	1479.4	2053.9	2639.3
$^{240}Pu_{eff}$ [percent]	6	14.8	23.3	31.4	39.3
$^{239}Pu_{eff}/^{240}Pu_{eff}$	15.6	5.75	3.3	2.18	1.55

5.2 MCNPX Results

The primary results of the simulations are the RADs constructed for each of the assayed fuel assemblies. As an example of how the shape and magnitude of the real part of the RAD may vary depending on the grade of the contained plutonium, Figure 9 compares RADs of 100W and 100R fuel assemblies. The magnitude of the fast and slow component displays a significant dependence based on the plutonium grade. Since RPu contains more ^{240}Pu than WPu, we expect more SF giving rise to higher fast component of RAD for 100R fuel assembly. Figure 9 also shows the ratio of these two RADs (right) where the ratio of RPu/WPu is smaller in the slow gate because of the parasitic absorption of thermal neutrons in the ^{240}Pu.

In order to track the relative changes between the fast and the slow components based on the plutonium isotopic composition, a ratio of the integrals of the RAD in early and late time domains needs to be analyzed. Table 5 lists the numerical values of individual simulated rates for an early gate from 2-8 μs and late gate of 70-90 μs for all five fuel assemblies listed in Table 4.

Table 5. Simulated Singles and Reals rates in early and late gate for full assemblies

Pu Grade	S (cps)	R gate (2-8 µs)	R gate (70-90 µs)	Ratio RPu/WPu
100W	10093	254.06	20.5	12.39
75W	20780	567.93	43.92	12.93
50W	31394	867.42	64.83	13.38
25W	41856	1156.17	83.04	13.92
0W	52119	1430.32	98.27	14.56

The ratio of the fast gate (2-8 µs) to the slow gate (70-90 µs) is sensitive to the WPu fraction when the pins are uniformly distributed. Figure 10 shows the fast/slow ratio for the case where the total mass of Pu stayed fixed at 6,024 g. The slow gate counts are reduced as the ^{240}Pu increases because of thermal-neutron absorption. However, the thermal neutrons are derived from the HDPE reflector so the effect is concentrated on the perimeter pins.

To evaluate the doubles response as a function of the plutonium mass and fuel grade, calculations were performed where the plutonium mass was varied from 1200 to 6000 g for both WPu and RPu. The ^{238}U was increased to keep the total weight of the assembly constant. Figure 11 shows the early gate (2-8 µs) doubles rate as a function of the total plutonium mass. As expected, we see that the RPu has a much higher doubles rate because of the high spontaneous fission rate per gram of plutonium.

The data shows a factor of ~5-6 difference between the WPu and RPu materials for the D rate. A direct substitution of RPu pins for WPu pins would be easily observed by the doubles that would increase because of the high spontaneous fission rate. However, if a small amount of RPu were to be added to a ^{238}U rod, the correct doubles rate could be maintained to mask the WPu removal. For this diversion case, we can measure the multiplication of the assembly that would be decreased because of the missing plutonium mass. The evaluation of the multiplication change will be evaluated later in this paper.

6. Doubles to Singles Ratios

For traditional neutron coincidence measurements, both the doubles and single rates are used in the analysis. The doubles rate is proportional to the spontaneous fission rate from $^{240}Pu_{eff}$ as well as the induced fission multiplication (M). The so called "known alpha" analysis[5] uses the D/S ratio and the known isotopic ratios to correct for the neutron multiplication. The D/S

Figure 9. The RAD for fuel assemblies with 100 percent reactor grade Pu (100R) and 100 percent weapons grade Pu (100W), and the ratio of the RPu/WPu on the right axis

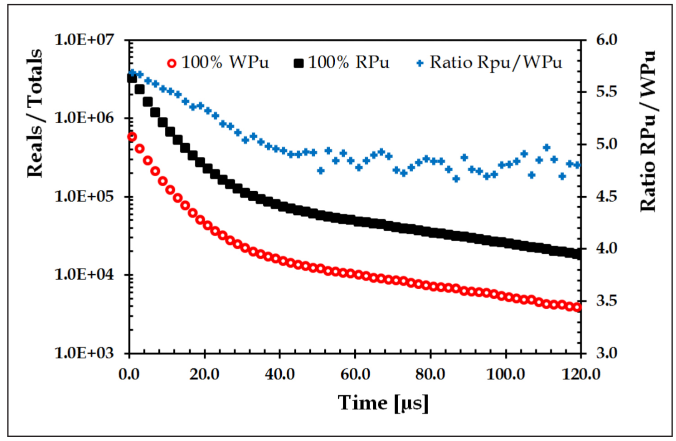
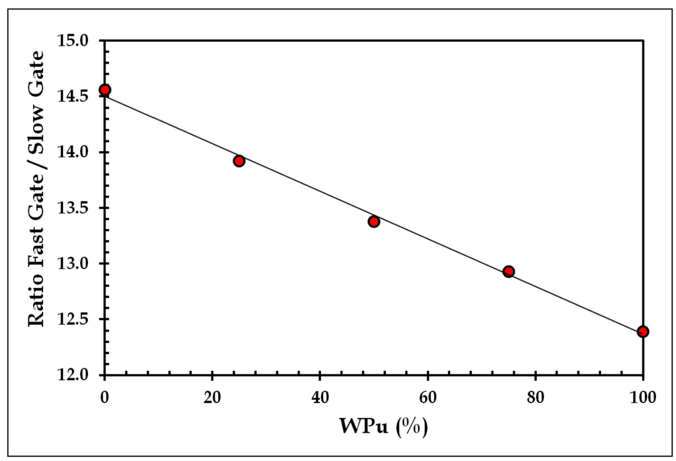


Figure 10. The ratio of the fast gate (2-8 µs) to the slow gate (70-90 µs) for fuel assemblies where the total Pu mass is held constant and the WPu was uniformly distributed



ratio has been used in the past[12] to obtain the multiplication of the assembly when the plutonium isotopic ratios are known.

The singles and doubles rates can be obtained from the equations

$$S = F \cdot \varepsilon \cdot M \cdot \nu_{s1}(1+\alpha), \text{ and}$$

$$D = \frac{F \cdot f_d \cdot \varepsilon^2 \cdot M^2}{2}\left[\nu_{s2} + (M-1)\frac{\nu_{s1} \cdot \nu_{i2}}{\nu_{i1}-1}(1+\alpha)\right] \quad (2)$$

where F is the spontaneous fission rate [fission/s·g], ε is the detector efficiency, M is the neutron leakage multiplication, α is the (α,n) to spontaneous fission neutron ratio, f_d is the doubles gate fraction, n_{s1}, n_{s2}, n_{s3} are the first, second, and

Figure 11. The ratio of the fast gate (2-8 μs) to the slow gate (70-90 μs) for fuel assemblies where the total Pu mass is held constant and the WPu was uniformly distributed

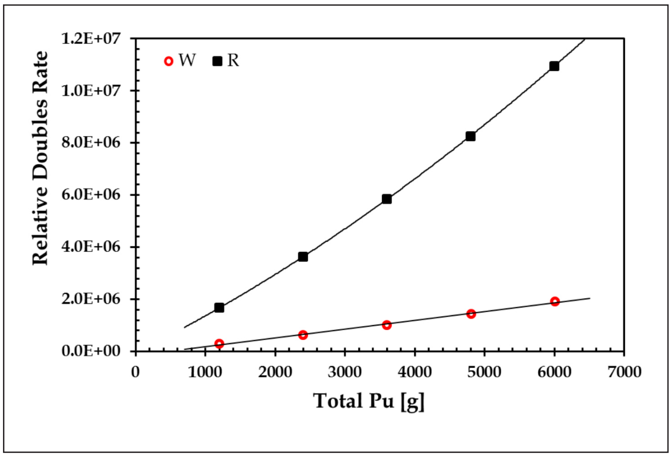
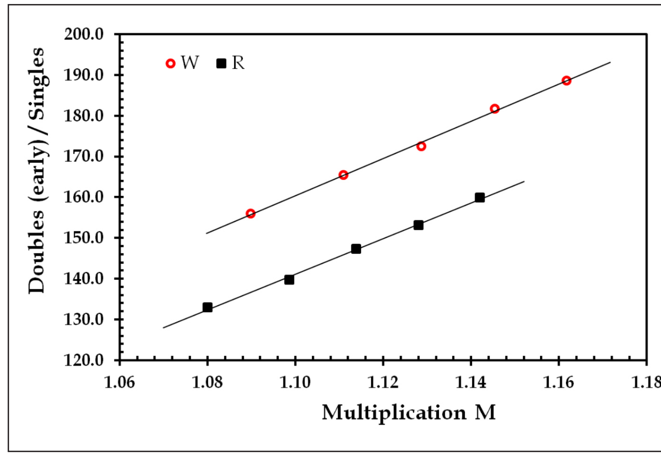


Figure 12. Calculations for the D/S ratio as a function of the M for a variable plutonium mass from 1,200-6,000 g



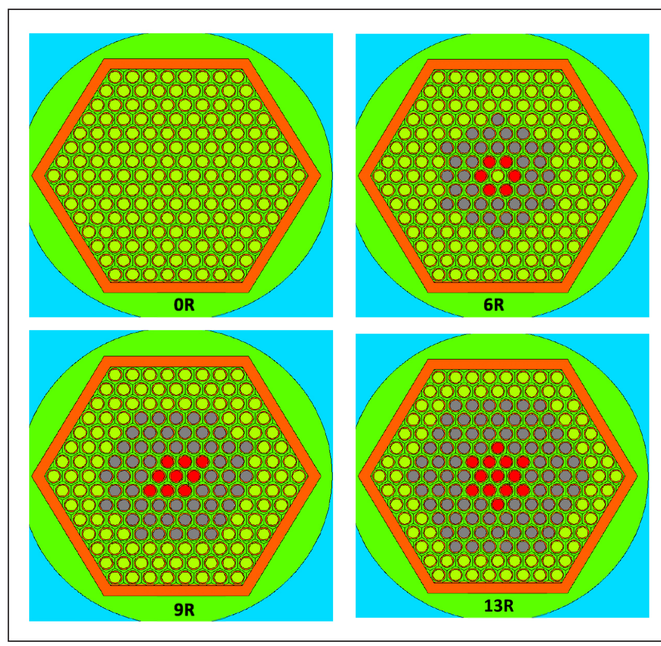
third reduced moments of the spontaneous fission neutron distribution, respectively, and n_{i1}, n_{i2}, n_{i3} are the first, second, and third reduced moments of the induced fission neutron distribution. Thus, the D/S ratio varies approximately M^2/M, so the ratio is proportional to M.

Figure 12 shows the D/S ratio as a function of M for WPu and RPu where the total plutonium mass was varied from 1200-6000 g. The total mass of the assembly was held constant by using ^{238}U to replace the plutonium. We see that the D/S ratio increases as the M increases and that the WPu has a higher D/S ratio than the RPu. The substitution of R grade Pu for W grade will decrease the measured multiplication as illustrated in Figure 12.

7. MCNPX Results for Penetrability of the Measurement for FBR-type Fuel Assemblies

We can assume that any diversion of WPu would be confined to the interior pins because the exterior pins can be verified by HRGS. However, the thermal-neutron albedo does not penetrate past the first two to three rows of pins. Thus, the presence of RPu in the interior is not picked up by the shape of the RAD distribution. However, the fast-neutron M is decreased as the WPu is removed and a lesser mass of RPu is substituted in the interior. Note that RPu has substantial fast-neutron multiplication. However, the passive neutron D and S rates from RPu are very high so that the number of pins for the diversion has to be reduced by a factor of 6-7 compared to the number of WPu pins removed. This reduction of the total Pu mass results in a decrease in M that is picked up by the known M analysis.

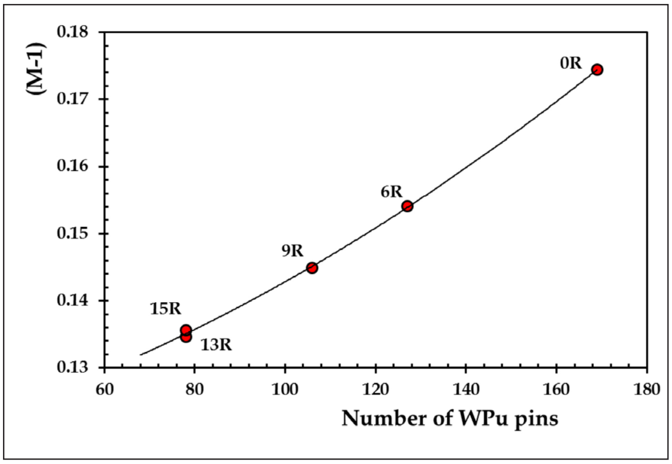
Figure 13. Fuel pin substitution scenarios where up to 78 WPu (yellow) pins are replaced by depleted uranium pins (gray) and RPu pins (red)



To evaluate the sensitivity of the known M method to the pin substitutions, we simulated several pin substitution scenarios. Figure 13 shows the pin geometries that were simulated with the MCNPX code. The inner pins were a combination of RPu pins (red) and depleted uranium pins (gray). The total number of pins (169) and the assembly mass were held constant, and the RPu pins were mixed in with the uranium pins.

The (M-1) values for the geometries depicted in Figure 13 are shown in Figure 14. The number of RPu pins was determined by the requirement that S rate be approximately the

Figure 14. The change in (M-1) as a function of the number of WPu pins where the substituted pins consist of depleted uranium plus 6, 9, 13, or 15 RPu pins. The statistical uncertainty of the (M-1) values as determined by MCNPX is a ~0.01 percent, therefore the associated error bars would be smaller than the size of the data symbols.



same as for the 100WPu case. We see that the M for the altered fuel assemblies decreases as the number of WPu pins decrease with depleted uranium replacement. The decrease is only slightly more when RPu pins are in the substitution mix because the number of RPu pins is limited to a small fraction of the uranium pins to keep the neutron emission rate to be consistent with 100WPu pins.

The falsification of the D and S rates to achieve the same M value as for 100WPu case would be very difficult, because when the RPu substitution pins reach the necessary D rate, the S rate is incorrect. To get the correct D/S ratio for the RPu substitution, it would be necessary to add a spontaneous fission source such as ^{244}Cm where the D/S emission ratio is higher than for ^{240}Pu. However, the ^{244}Cm source has an 18.1-year half-life so the correct mass to be added to the assembly would decay to an incorrect value in a couple of years. This change with decay time would primarily apply to fuel in storage waiting for use in the reactor.

Of key importance is that the neutron detector has the capability to change M by the addition or removal of the Cd liner, so that the correct D/S ratio for one configuration would be incorrect for the other configuration.

The sensitivity of the M-1 measurement would be dependent on the detector design as well as the control of the singles background during the measurement. For the FBR MOX assemblies, the singles signal to background ratio is expected to be high, so the error in the background subtraction would become small. The doubles statistical error would be ~0.6 percent in 10 min so the ~30 percent total variation for (M-1) shown in Figure 14 for 91 rods substituted (15 RPu + 78 DU) would be easy to observe. For a single rod substitution the change would be ~0.3 percent, so the observation would be impossible at the 3 sigma level. On the other hand, removal of about five to seven rods may be close to detection limit. Making a more quantitative estimate of the sensitivity is the subject for future proposed work.

8. Triples Rates Verification

The capability of a diverter to falsify the M via artificial changes in the D and S counting rates can also be blocked by doing a multiplicity analysis that includes S, D, and *T* where *T* is the *Triples* Rate. For detectors with typical die-away times, the error related to *T* gets to be large because of the high Singles rates that result in a large error from the high accidentals rates. We have mitigated this problem by reducing the die-away time for the detector. To estimate the error in the plutonium mass measurement using the current detector design, we used Ennslin's Figure-of-Merit code[13] that gives an error in the *T* rate of 4.2 percent for a 15-minute measurement for the present design illustrated in Figure 8.

The error in the *T* rate based on the present detector design is higher than what is necessary for typical applications.[14] The mini-epithermal neutron multiplicity counter (mini-ENMC)[15] shown in Figure 15 has a measured efficiency of 63 percent and a die-away time of 19 µs. If the FBR assembly were placed inside the mini-ENMC, the error for the *T* rate would reduce to ~0.33 percent and 0.07 percent for the D rate with a 15-minute measurement. However, a much smaller and more portable detector can be designed for the verification of FBR fuel assemblies. A detector with the same approximate dimensions as illustrated in Figure 8 could be applied to the FBR verification problem.

For use of the *T* rates, we would redesign the detector head of the same size as illustrated in Figure 8, but to have a higher efficiency (~ 20 percent) and a longer die-away time. For this type design, the FOM code predicts an error of ~1 percent in the measured *T* rate in a 15 min measurement.

The various diversion scenarios described above for the removal of WPu pins cannot satisfy the multiplicity observables of the S, D, and *T* rates by the insertion of pins containing ^{241}Am and SF isotopes such as ^{244}Cm. The actual S, D, and *T* rates depend on the first, second, and third moments of the plutonium isotopes that cannot be reproduced by any combination of substitution isotopes.

Figure 15. The high efficiency mini-ENMC with split geometry for inserting plutonium samples or FBR assemblies

Summary

Different types of MOX FBR assemblies and LWR fuel assemblies have been measured for several decades using neutron coincidence methods. The standard analysis methods have determined the $^{240}Pu_{eff}$, the M, and the total plutonium by separately determining the plutonium isotopic ratios. The detectors have been lined with Cd to prevent the thermal-neutron albedo return from the detector moderator to limit the multiplication to fast-neutron multiplication. In the present paper, we have added the thermal-neutron return multiplication that can be compared with the fast-neutron multiplication when a Cd liner is returned to the wall of the sample cavity.

We have investigated three enhancements for the analysis methods to standard coincidence counting to prevent the masking of WPu pin removal and substitutions. These enhancements are:

1. The known M with Cd ratio measurements,
2. The DDSI RAD shape analysis, and
3. The multiplicity measurement of S, D, and *T*.

We found that the DDSI enhancement lacked penetrability if all of the diverted pins are located in the interior region of the assembly, so we have not pursued the approach for diversion cases. However, the "known M" analysis combined with the Cd ratio measurement would be effective in blocking the removal of WPu because the correct M cannot be falsified for the two detector configurations and there is penetration to the interior of the assemblies.

The most elegant method to prevent the masking of the diversion of WPu pins is the use of the multiplicity observables of S, D, and *T*. We note that such a measurement is based on SF and fast-neutron fission and has penetrability to all regions of the assembly. The sensitivity of this method needs to be quantified for different diversion pin patterns in future work.

It is important that methods 1) and 3) above are currently supported by the INCC analysis code that is routine use by inspectorates world-wide. The data collection of S, D, and *T* is identical for both analysis methods. The primary focus of the present paper was to show the capability of verifying that the $^{240}Pu_{eff}$ fraction in the plutonium corresponds to WPu. The method needs to be independent of the position of the rods in the assembly. This is of key interest in the nuclear disarmament to verify that the plutonium is weapons grade versus reactor grade.

The "known M" compared to the "known alpha" verification method was actually applied by a Russian IAEA inspector (V. Risikov) during an inspection visit to the BOR-60 FBR reactor in Russia in 1988[16]. In that case, the Pu isotopic ratios were accidently mis-declared by the operator, and there was a $^{238}Pu/^{239}Pu$ ratio discrepancy compared with the HRGS measurements. The known M analysis indicated that the ^{238}Pu fraction had been mis-declared, and a subsequent investigation showed that the ^{238}Pu fraction had been declared to be lower than it actually was. When the correct isotopic ratios were used, the neutron measurement gave agreement between the known alpha and known M analyses paths.

A measurement of the plutonium isotopic ratios for the exterior pins using HRGS can provide the $^{238}Pu/^{239}Pu$ and the $^{240}Pu/^{239}Pu$ fractions for the sub-assembly. The neutron based $^{240}Pu/^{239}Pu$ fraction can show the uniformity of the plutonium for both the outer and inner regions. This combined neutron and gamma capability can provide the verification of the WPu mass including the ^{238}Pu fraction for the complete assembly regardless of the pin distribution. This capability to verify a full MOX assembly for WPu mitigates the need for scanning individual pins prior to loading the assembly. The pin scanning introduces many questions as to the continuity of knowledge for the pin handling and the location of the pins in the final assembly. The portability of the neutron coincidence detector also allows for measurements of assemblies during the initial loading (real time), as well as during subsequent transport and storage.

Acknowledgements
This work was funded in part by the U.S. Department of Energy (DOE), National Nuclear Security Administration (NNSA), Office of Nonproliferation and International Security.

Keywords: neutron detector, coincidence counting, fast breeder fuel assembly, plutonium measurements

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Book Review

By Mark L. Maiello, PhD
Book Review Editor

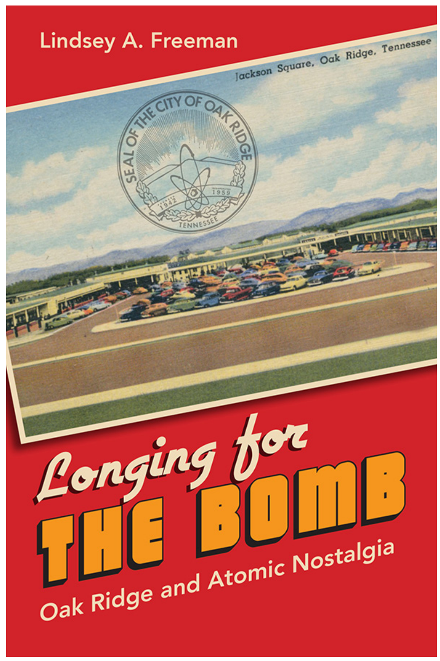
Longing for the Bomb Oak Ridge and Atomic Nostalgia

Lindsay A. Freeman

Softcover, 238 pages
ISBN 978-1-4696-2237-8
The University of North Carolina Press, Chapel Hill, 2015

The bomb making years of the '40s and '50s have left a legacy of literature, a legacy of historical personalities and a legacy of place. The latter are the facilities that were the birthplaces of the bombs and of the nuclear materials at the hearts of those weapons. This book, at once a personal experience and simultaneously a detached treatment of a purpose-built society, is a wonderfully written guide through the city of Oak Ridge, Tennessee, the secret, guarded, enclosed community whose residents created (unbeknownst to many of them) the enriched uranium for the Hiroshima bomb.

The author, whose family hails from Oak Ridge, was able to tap into family memories in order to balance and gauge her research findings. And, what she found has resulted in a deeply textured and nuanced narrative, easily accessible to any reader. This is in reality, a sociologic treatment that reads like a homespun story: factual, meticulously researched but presented without pretense. It contains a very personal thread because of the author's ties to the community. Conversations with her grandmother float to the surface at opportune moments to



reinforce and personalize her research. Her work reveals both the highs and the lows of living in a pre-planned government community — very socialistic in certain ways — that fostered a paranoid culture of secrecy while attempting to keep its citizens both happy and engaged in a project whose outcome for security reasons, many knew very little of. Though the community succeeded in its ultimate goal, there were ethical and moral failures at other levels — the most egregious being reflections of American culture at that time.

Despite her allegiance to Oak Ridge, one senses after reading a few pages that the author is quite rightly allied to her research allowing her to maintain objectivity. She pulls no punches when it comes to negative criticism. For example, she begins her story with the government's 1942 appropriation of private land that effectively wiped away five towns so that Oak Ridge could be built. Were the land owners properly compensated? Freeman answers no, especially when one considers that these properties had been held by families for generations. We are then treated to her study of the planning, building and running of one of the nation's first designed societies — the forerunner of the suburban Levittowns that would arise with the U.S. standard of living after the end of the war.

Freeman's view of life in a military-ruled community engineered not necessarily with democratic principles at the forefront, but with a secret national objective in mind, is wide ranging. She describes the influences the military had on the everyday life of the laborers, scientists, families, administrators and other support staff that made Oak Ridge their home. Freeman comments on the effects of gender differences in the labor force, the class stratification of families based on the professional credentials of the male breadwinners, the racial segregation that was designed into the town plan, the secrecy and surveillance that encouraged the reporting of potential spies, the many entertainments, services and privileges afforded most residents to keep them from visiting nearby towns and perhaps "talking too much," and their reactions to the horrific maiming, fatalities and physical destruction their mission goal (once revealed), had caused. It sounds like a dystopian culture but in fact, its residents — its white residents at least — found it from their limited perspective, enjoyable, comfortable, patriotic and even exciting. It was unique and it allowed people to serve their country albeit in a very circumscribed and knowledge-fragmented manner. Its African-American citizens saw it as a chance for work at higher wages, a decent place to live, and a way to exhibit their patriotism but subject to the same atrocious, inhuman, segregationist tactics that befell the American society of the day.

Freeman frames the entire narrative in the context of nostalgia — specifically "technological or scientific nostalgia" — the current passion for what America built or discovered in the past and the tourism that has developed for these ventures. She indicates the manner in which the historical narrative fed to curious tourists can subvert the true nature of the past they seek. Tour guides — often an elderly former worker of the facility — espouse their own partisan viewpoint that has been cleansed of opposing perspectives. Many museums devoted to preserving the history of these legacy sites often sanitize the message or at least focus it such that the negative repercussions of the technology or the discovery are marginalized or not discussed at all.

The author contends that the current inhabitants of Oak Ridge do more or less the same. Take for example their view of segregation and African Americans. In the past, African-American Oak Ridgers were given the most dangerous and least desirable jobs. Their housing was substandard and separated from the neighborhoods of the white professional classes — Oak Ridge's "other side of the railroad tracks." It was planned segregation. In the early days of the city, there was also no housing for married black couples. Sex became clandestine. Treated unfairly in accommodations, employment, and even in marriage, they also fell victim to Oak Ridge's military secrecy in a most heinous manner. Medical experiments to determine the toxicity of plutonium were conducted at Oak Ridge on African Americans without their consent — a despicable crime perpetrated by the veil of concealment the hung over the community. Segregation, unfair practices, medical crimes — none are spoken of in detail today. The author contends they are acknowledged only fleetingly in the context of the "fevered wartime explanation" which simply stated was: There was a mission to accomplish: build the bomb — win the war. In short, the means justified the end.

Also pervading Freeman's study is the concept of atomic utopianism. The ideal pre-planned community was not a new idea in the 1940s. But, the notion that a crucial mandate actually existed for such a community was an exciting idea. The very survival of the United States was thought to depend on it. The idea of contributing to a mission-driven community that combined massive industrial infrastructure with newly minted housing and supporting facilities placed into an idyllic setting in the Appalachian Mountains was appealing to planners, engineers and architects. Almost stealing a theme from science fiction (or is it the other way around?), Oak Ridge was where the new science of atomic physics could reach its apex and in the process, save the nation. In the literature of science fiction, dystopian communities often masquerade as paradises. The more sophisticated story lines incorporate a legitimate mission for these communities that front an evil, all-controlling government. Often these fictitious places are refuges from humanity's tragedies and mistakes. They can be hermetically separated from the environmental disasters that humanity visits on itself. *Longing for the Bomb* can be viewed as a real-life critique of the fictional dystopias depicted in novels like *Logan's Run* or motion pictures like *Aeon Flux* — but without the extreme disasters that spawned these invented, closed societies. Like the domed and walled off cities of these fantasies, Oak Ridge was for a privileged many, a grand place to live — but not so for all because American cultural norms were not challenged — they were sustained; very telling for a federally built society. Another example: gender separation in the workplace — what we would term today as sex discrimination. The most famous example of this is the "Calutron girls" who operated their controls for the enrichment of U-235 without actually knowing what they were making. The reasoning for employing young women was simple. Girls were thought to be better suited than men to the repetitive work required.

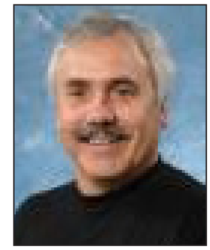
Oak Ridge's metaphorical dome was the controlled ingress and egress of its citizens. Background checks were performed on each before arrival. Information about the city's mission was fractionated so that only a few knew the ultimate objective. The word "uranium" could not even be spoken. Loose lips were punished by unemployment and even imprisonment. Government informers listened and watched for violators. Entertainment such as bowling allies, ball fields, an Oak Ridge Symphony Orchestra, the largest swimming pool in the South, and seven movie theaters kept the inhabitants inside the guarded perimeter of the community to not only curtail the flow of information to the outside but also to make it harder for the enemy to gauge the size and the specifics of the project. And all the while, housing, schools, free transportation, a greenbelt of parks and other amenities were provided by the government. All were inducements to come to Oak Ridge and to stay for the duration.

That all said, the author balances the idealism with the negative feelings Oak Ridgers felt about security, housing, and the caste system that developed around the most needed citizens (the physicists), and everyone else. This culminated in their conflicted feelings once they knew of their contribution to the atomic bomb. With the dropping of the Hiroshima bomb on August 6, 1945, the secret world of Oak Ridge ended. The media and the military revealed that the "terror bomb" had been unleashed. Those that had no knowledge of the city's mission were at once perplexed about the outcome of their button-pushing and dial-turning, amazed that their contributions produced such a devastating weapon, and celebratory over the achievement and the anticipated end of the war. But others, the scientists with the knowledge of what was being produced, were conflicted and in some cases disgusted by the outcome. They questioned the dual bombing of Japan. Why had we not given Japan more time to surrender? Others worried about the impact atomic warfare would have on the future of world society.

Oak Ridge was the first "scientific community." It regards itself as the home of nuclear science to this day. The atom remains its symbol and the word "atomic" still adorns many a business in Oak Ridge. A two-day Secret City Festival honors its atomic heritage each year. But present day Oak Ridge reveals another layer of atomic nostalgia through the act of tourism. It seems to Freeman that today, science walks hand-in-hand with tourism in a less than empirical way. Tour guides repeatedly remind you of the past "miracle" accomplishments of Oak Ridge. The author, on one of the many bus tours now available, experienced an overenthusiastic guide trying to connect science and religion through (here it is again), science-fiction – via the 1960s TV show *Star Trek*. "Beam me up Scotty" (never spoken in the original *Star Trek* despite the author's assurance to the opposite) was used by the guide in a surreal attempt to invoke a divine connection with the city's scientific accomplishment. Such is the layered, nuanced, slightly skewed nature of current atomic tourism and scientific nostalgia that Freeman eloquently discusses.

The book is a vivid, wide ranging journey into a very interesting corner of U.S. atomic history — a place that was itself a remarkable achievement; a place that helped produce a world-changing achievement, but also a place that was terribly flawed by the mid-twentieth century American culture that spawned it. Freeman's discourse is brimming with accounts of individuals who lived the Oak Ridge experience. She deftly utilizes their written words taken from military speeches, newspaper stories and other sources. The town's many cultural flavors are brought to the table in a manner easily consumed, understood and enjoyed. There is much to praise here including the author's honest, factual, scientific investigation rooted in personal interest and personal connections but objectively discussed and explained. It is a book deserving of attention far beyond the borders (or fences) of its subject.

Taking the Long View in a Time of Great Uncertainty



A View from the International Community

By Jack Jekowski
Industry News Editor and Chair of the Strategic Planning Committee
and

L. David Lambert, Special Guest Columnist,
Past Chair of the INMM Nuclear Security and Physical Protection Division [1]

"It takes a very long time to become young."
~ Pablo Picasso

In this issue's column I am delighted to present a perspective on the future from the international community, provided by a long-term Senior Member of the INMM, L. David Lambert. After retiring from Oak Ridge National Laboratory, David has spent the last four years "living" international service, first in Vienna at the International Atomic Energy Agency (IAEA), and more recently supporting the Defense Threat Reduction Agency in the Republic of Kazakhstan for the contractor, Gregg Protection Services/Centerra Group.

As we have reflected upon in previous columns,[2] we are rapidly approaching a critical moment in the evolution of "all things nuclear" with respect to the preparation of the next generation to accept the responsibilities associated with the legacy we are leaving them. David's insights below once again call for an urgent response to this global issue, one that the Institute undertook more than a decade and a half ago with the engagement of universities that has led to the formation of Student Chapters worldwide. As the Institute's new operational strategies for the coming five years are revealed at this year's Annual Meeting in Atlanta, we will all better understand the challenges that lie ahead, and the responsibilities those of us in the "older generation" have for ensuring a safe and secure future for our world.

A View from the International Community

Baby Boomers. Generation X. Millennials. These are recognizable, common terms used to describe people from certain time periods of the world's history.[3] Many stories have been written about the technology changes that have occurred in just the last fifty years. We sometimes find those stories whimsical and amusing … and perhaps a little disconcerting at the same time, depending on age group. What is a trip down memory lane for someone in their 60s is more than a bit confusing for a teenager. Something as simple as a headlight dimmer switch on the floorboard of an old Ford Fairlane is incomprehensible to someone today who has always dimmed their headlights by pulling the handle on the left side of the steering column. Figuring out the latest smartphones, tablets, and video games is just as confusing to many in their 60s or 70s, but child's play (pun intended) for that teenager today.

Even as debates continue about nuclear weapons programs, development of new nuclear power programs, handling and storage of spent nuclear fuel, and the ever-increasing application of nuclear medicine, one common thread links them — the need for competent people who accept responsibility for the security and accountability of nuclear technology and radioactive materials.

Countries with what are considered mature nuclear weapons programs prac-

This column is intended to serve as a forum to present and discuss current strategic issues impacting the Institute of Nuclear Materials Management in the furtherance of its mission. The views expressed by the author are not necessarily endorsed by the Institute, but are intended to stimulate and encourage JNMM *readers to actively participate in strategic discussions. Please provide your thoughts and ideas to the Institute's leadership on these and other issues of importance. With your feedback we hope to create an environment of open dialogue, addressing the critical uncertainties that lie ahead for the world, and identify the possible paths to the future based on those uncertainties that can be influenced by the Institute. Jack Jekowski can be contacted at jpjekowski@aol.com.*

tice stockpile stewardship, but are often are faced with significantly less manufacturing expertise than existed in their formative years. Experienced people with institutional knowledge of the practices, processes, and strategies for protection of those dangerous materials, in all their various forms, continue to disappear with each passing year. Lessons that could be learned, practices that are still viable, ideas whose time has come — all gather dust and are either ignored or forgotten as the next generation moves on to their own set of priorities.

A good example from the International Atomic Energy Agency (IAEA) is what happened as support for new nuclear power programs began to grow. Countries with existing nuclear power plants (NPP), or those that had an NPP in the recent past, were not considered "new" programs. What became apparent over time is that some countries with existing NPPs knew how to operate them, but the experienced people with institutional knowledge about designing, constructing, setting up, and initially licensing a new NPP had disappeared. The same loss of knowledge happened in countries that shut down a previously operating NPP, in many cases over a shorter period of time as the people with such knowledge migrated to other countries, leaving a legacy of stored nuclear materials without the expertise to properly manage them for decades into the future.

In the safeguards and security areas that are important to the INMM, technology to protect and manage nuclear materials continues to increase at a furious rate, perhaps too fast for many less-developed countries. Threats applicable to the theft, diversion, or sabotage of nuclear materials or their associated facilities seem to increase in complexity and motivation, with malevolent acts making headlines almost weekly. All of this is happening as a new generation of women and men are taking leadership roles as technical experts and managers in the so-called "nuclear renaissance" era. One tool the INMM has in its arsenal of global support to fill that pipeline of the new generation is the knowledge and experience of its diverse membership that can be shared with that new generation, easing the transition.

Since its inception, the INMM has been leading the charge to educate anyone who would listen about the nuclear world, provide a platform for new technologies and methodologies, and focus the capabilities of subject matter experts from all over the world on vital issues. Members of the Institute continue to speak and demonstrate the value of their expertise and experience, and the INMM has enhanced its ability to communicate by taking steps to make its voice younger.

With twenty student chapters around the globe (and more on the way), the INMM has made great strides in the last several years to provide those student chapters with opportunities to learn and grow by encouraging their participation in Executive Committee meetings and Institute activities. Members can volunteer to become mentors to less experienced individuals through the website and technical divisions. The INMM's professional chapters also provide a mechanism for those students who are seeking to learn more about their specific areas of interest through the Institute's Annual Meeting, technical workshops, and other venues. Enhancing the knowledge of students helps each of them individually, but also their sponsoring faculty and universities collectively. These efforts have been significantly enhanced in recent years by the engagement of the U.S. Department of State's Partnership for Nuclear Security (PNS) initiative,[4] which has sponsored several new international student and professional chapters, facilitating their engagement with the INMM.

But just being exposed to knowledge and experience is not enough for our students. Stronger identification with the INMM is needed by all members, manifested by more active involvement in Technical Division events. Visible, tangible support of the Technical Divisions for student activities and events raises the interest level of those who want to be involved in something that is dynamic, fresh, and thought-provoking. Increased involvement in the Technical Divisions means a greater population of potential mentors that can impart that knowledge and experience. Highly experienced mentors have to be willing, and supported, to advise entire student chapters, and *their sponsoring faculty.*

The International Nuclear Security Education Network (INSEN),[5] supported in part by the IAEA, is another avenue that the INMM and its student chapters should be exploring. An entire nuclear security curriculum, with teaching materials and some textbooks, has been created by INSEN to teach faculty members of interested universities about all aspects of nuclear security through professional development courses (PDC) funded primarily by the United Kingdom (U.K.). Because the INMM provides online tutorials and conducts training/workshops on the entire spectrum of nuclear security subjects through its Technical Divisions, partnering with INSEN would be mutually beneficial. Universities with INMM student chapters could be encouraged to engage with INSEN, and the INMM might, in turn, be able to inspire more INSEN members to develop INMM student chapters.

INMM's role in the creation of the World Institute of Nuclear Security (WINS)[6] with its important workshops, publications, and the WINS Academy is yet another example of how we can ensure the future safety and security of the nuclear world, as are opportunities with numerous other organizations, such as the ASIS Certified Protection Professional (CPP) and Physical Security Professional (PSP) programs;[7] the National Skill Academy,[8] Nuclear; and, the U.S. Department of Energy's National Training Center.[9]

The future is upon us. The INMM has incredible resources to involve, encourage, and empower current and future professionals who protect and safeguard assets in the nuclear industries. The Institute has a global membership to lead, direct, and mentor others to continually learn and improve through meetings; workshops; professional and student chapters; and Technical Divisions that continually plant the seeds of knowledge and responsibility that will germinate and spread to ensure a brighter future world.

The INMM owns an important realization, "If not us, who; and if not now, when?"

"A sword, a spade, and a thought should never be allowed to rust."
~James Stephen

Endnotes

1. David Lambert is on assignment in the Republic of Kazakhstan with Gregg Protection Services/Centerra Group, managing training activities for URS Federal Services International and the Defense Threat Reduction Agency (DTRA) in support of nonproliferation, nuclear security and human resource development projects for the Global Nuclear Security (GNS) Program. David previously served for two years as the Senior Nuclear Security Training Officer with the International Atomic Energy Agency in Vienna, having retired from a thirty year career at Oak Ridge National Laboratory where he was Deputy Director for Program Development in the Global Security and Nonproliferation Program Office. David is a Senior Member of the INMM and was Chair of the Nuclear Security and Physical Protection Technical Division for several years. He authored the *Best Practice Guide on Human Reliability as a Factor in Nuclear Security* for WINS. The perspectives he provides in this article are his personal observations of the international nuclear sector based on more than four years of experience.
2. See past columns in the *Journal of Nuclear Materials Management (JNMM)* that speak directly to this "next generation" issue, including: *JNMM*, Summer 2014, Vol. XLII, No. 4, "*Taking the Long View: Throwing Down the Gauntlet to the Next Generation of Nuclear Stewards – the Enduring Nuclear Legacy*," pp. 86-89; ibid, Fall, 2011, Vol. XL, No. 1, "*Taking the Long View in a Time of Great Uncertainty: The Changing Face of INMM at the 52nd Annual Meeting*," pp. 56-57; and ibid. Fall, 2014, Vol. XLIII, No. 1, "*Taking the Long Veiw in a Time of Great Uncertainty: Turning the Corner*," pp. 65-67.
3. It should be noted that these terms are generally considered to be U.S.-centric, however, they are also used in other countries, although the timeframe for distinguishing them, and their relative demographic impacts may be different. See, for example, "Generations and Geography: Understanding the Diversity of Generations around the Globe," http://www.tammyerickson.com/images/uploads/TEA_Generations-and-Geography.pdf; and data such as http://ec.europa.eu/eurostat/statistics-explained/index.php/Population_structure_and_ageing. Also see http://www.tomorrowtodayglobal.com/2010/12/07/uk-and-us-baby-boomers-are-not-the-same-so-please-stop-generalising-generations/ for a discussion of how the Post-WWII world of the United Kingdom, and Europe, in general, postponed the "Baby Boomer" shaping of society for several years.
4. See https://www.pns-state.net/en-us/
5. See http://www-ns.iaea.org/security/workshops/insen-wshop.asp
6. See https://www.wins.org/
7. See https://www.asisonline.org/Certification/Pages/default.aspx
8. See https://www.nsan.co.uk/
9. See https://ntc.doe.gov/

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Kobe Portopia Hotel
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December 5–9, 2016
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Vienna, Austria

January 10–12, 2017
32nd INMM Spent Fuel Management Seminar
Washington Marriott Georgetown
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July 16–20, 2017
INMM 58th Annual Meeting
Renaissance Indian Wells
Indian Wells, California USA

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Papers are acknowledged upon receipt and are submitted promptly for review and evaluation. Generally, the corresponding author is notified within ninety days of submission of the original paper whether the paper is accepted, rejected, or subject to revision.

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