U.S.NRC
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Protecting People and the Environment

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26TH ANNUAL REGULATORY INFORMATION CONFERENCE

ANSWERS TO UNANSWERED QUESTIONS FOR RIC 2014

Tuesday, March 11, 2014

EDO Remarks

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OPENING SESSION
Tuesday, March 11, 2014, 8:30 a.m. – 10:00 a.m.

"An Overview of NRC Operations and Looking Ahead" – Remarks by Mark Satorius, NRC's Executive Director for Operations

The questions below were not answered during the above session.

QUESTION 1: What specifically are you doing to establish greater consistency and regulatory certainty between all regions?

ANSWER 1: This question was addressed during the RIC session titled, "Regional Session – Contemporary Nuclear Power Plant/Regulatory Issues," and a video is available on the NRC website.

QUESTION 2: Among challenges going forward, you cited "continuing adaptation of regulatory framework." How does this relate to the Chairman's comments regarding regulatory uncertainty?

ANSWER 2: While changes to the regulatory framework could be a departure from the status quo, any changes made by the agency will be done in the spirit of increased certainty, openness, clarity, and efficiency. These Principles of Good Regulation will help us ensure appropriate evolution of the regulatory framework over time is done in a responsible manner. This is consistent with the Chairman's message of continuous learning and making decisive changes when we have the information that supports them.

QUESTION 3: With the experience level of the agency's staff being challenged, how do you ensure the Principles of Good Regulation are maintained (e.g., new interpretations of long-standing issues/rule/regulations are minimized)?

ANSWER 3: The agency continues to work hard at knowledge management and ensuring new employees are given the training, tools, and resources they need to be effective regulators. The agency and industry are constantly learning new things about both technical issues and regulatory approaches. We must do our due diligence to evaluate new issues, ensure our existing regulatory framework provides adequate protection, and make changes as appropriate and needed. If new information causes us to consider different interpretations of existing rules, they should be evaluated using a structured, predictable process and decisively resolved.

QUESTION 4: Why have the staff recommendations on foreign ownership, control, and domination been delayed? When can we expect this report to be issued and will it be available to the public?

ANSWER 4: The staff is continuing to review viable technical, legal, and procedural policy options to provide to the Commission. The staff is currently working to provide this paper in the summer.

QUESTION 5: With the spent fuel pool study showing no significant safety improvement from expedited spent fuel pool offload, does the NRC consider the question closed?

ANSWER 5: The issue of expedited transfer of spent fuel is currently pending before the Commission. The NRC staff will carry out whatever direction the Commission provides in its Staff Requirements Memorandum.

QUESTION 6: With NRC budget limitations (equivalent to 2007), how will the numerous public outreach meetings for rulemaking activities (e.g., Waste Confidence) continue to be funded? Will these come from fees paid to NRC by licensees and hence expect higher fees in the future? Have "meaningful" comments come from these meetings to justify their costs?

ANSWER 6: While the NRC budget has been essentially flat over the last several years, the NRC budget has, and will continue to, include the necessary resources to support public outreach for rulemaking activities. The NRC considers public involvement in, and information about, our activities to be a cornerstone of strong, fair regulation of the nuclear industry. We recognize the public's interest in the proper regulation of nuclear activities and provide opportunities for citizens to be heard. The NRC will continue to collect fees in accordance with the Omnibus Budget Reconciliation Act of 1990 (OBRA 90) which requires the NRC to collect approximately 90% of its budget in the year appropriated through fees from its licensees. Annual fees are billed to the classes of NRC licensees to collect their recoverable budget, not collected from fees for services. The number of public meetings budgeted to support public outreach does not impact the collection of fees in accordance with OBRA 90.

QUESTION 7: You note that the NRC budget has been essentially flat, yet the NRC has not requested more (or additional) funding. What would NRC do with additional funding? Is additional funding necessary?

ANSWER 7: While the NRC budget has been essentially flat over the last several years, the NRC requested and received a Congressional reprogramming in fiscal year (FY) 2013 in the amount of $38 million. Congressional approval of this request gave the NRC access to prior year unused funding, thereby increasing the funding available in FY 2013. These resources supported an integrated prioritization of emergent and unfunded work and mitigated some of impacts to NRC programs as a result of the government wide budget reductions known as sequestration. In FY 2014, the NRC received its full budget request late in the fiscal year (March 2014) and is working to allocate those resources to our ongoing programs as well as to address emergent needs.

QUESTION 8: What changes, if any, are needed to current regulatory structure to make SMR licensing feasible?

ANSWER 8: The NRC staff has made significant progress in evaluating the regulatory framework to support the reviews of these new designs and has issued a number of information papers outlining how it plans to address various policy and technical issues related to Small Modular Reactors (SMR). The results of this evaluation indicate that no changes are needed to the current regulatory structure to make SMR licensing feasible. Some policy issues raised by the industry for SMRs are particularly challenging (e.g., emergency preparedness, control room staffing, physical security, annual fees, etc.) and will require extensive interactions with a number of external stakeholders. Additionally, some of the SMR designs will be first-of-a-kind reviews for the NRC staff and will have anticipated technical challenges such as the issue of "deeply embedded structures." However, for light-water SMR designs, there is sufficient guidance in place for the industry to propose exemptions, if desired, for these issues.

QUESTION 9: Can you provide examples of NRC participation in the licensing of new reactors in other countries, especially China, and any lessons-learned?

ANSWER 9: As part of the Multinational Design Evaluation Program (MDEP) [www.oecd-nea.org/mdep], the NRC shares information about design reviews with other countries. It is important to note that one tenet of MDEP is that each national regulatory authority retains the rights and responsibility for licensing new reactors in their country. Sharing information pursuant to MDEP activities is intended to make each country's review of new reactors more safety focused. The NRC shares information about several new reactor designs with other countries through participation in the EPR Working Group (WG), the AP1000 WG, the ABWR WG, and the APR 1400 WG.

China and U.S. are members of the AP1000 WG. The WG cooperates in the areas of: design changes to Chinese (Sanmen and Haiyang sites) and U.S. AP1000s (Vogtle and Summer sites); sharing information and experience on vendor issues; inspector exchanges; and support to the Canadian regulator for their Phase 3 review, when necessary. Other countries involved in the AP1000 WG include Canada, Sweden, and the United Kingdom.

The NRC staff completed two lessons-learned reviews in 2013 assessing the Part 52 licensing process and post-combined license implementation. Through these reviews, the staff identified several areas that would benefit from enhancements, and is currently engaging with industry, as well as with other external stakeholders, to effectively and efficiently implement improvements to the process. The Part 52 Implementation Lessons-Learned (1-Year Post Combined Operating License Issuance) was completed in July 2013 and is public (ADAMS ML13196A403).

QUESTION 10: If the budget is shrinking, how is it that the number of employees has nearly doubled in the last 10 years and add another 66 people this year?

QUESTION 11: NRC has about 1,000 more personnel than just a few years ago, but 4 units have shut down and more will occur near term. Why does NRC need more staffing?

ANSWERS 10 and 11: The NRC's FY 2014 Full Time Equivalents (FTE) is 3,752, compared to 3,059 FTE in FY 2004. Between FY 2004 and FY 2014, the agency budget for FTE increased by approximately 23 percent. The budget in FY 2015 includes 3,819 FTE's, which is essentially flat from FY 2014. The increase of 66 FTE in FY 2015 is attributable to a decrease in FTE's that occurred in FY 2014 to accommodate other required programs and priorities, such as the Integrated University Program, which was mandated through the FY 2014 appropriation. This reduction has not been reflected in FY 2015; however, the actual FTE needs in FY 2014 are comparable to FY 2015, and the NRC is continuing to assess appropriate staffing levels.

Workload at the NRC continues to shift and change. While the number of operating plants has decreased, NRC staffing has shifted internally as priorities change and skill sets are better aligned to support many changing important activities and efforts. The FY 2015 budget supports Fukushima Tier I and II activities, specifically increasing for reviews related to Mitigating Strategies; increasing cyber security licensing activities; increasing work related to Generic Issue-191; reviewing new applications for Medical Isotope production facilities; completing decommissioning activities at Kewaunee, Crystal River 3, and San Onofre Units 1 and 2; reviewing a new uranium enrichment plant license application; increases to review a possible amendment to expand operations for international isotopes; and progressing with revisions to the Fuel Cycle Oversight program as well as continuing support for new reactor licensing and construction inspection workload and the associated infrastructure.

As the NRC moves toward the future, it is unlikely there will be any growth over the next several years. In response, the NRC has adjusted its human capital strategies to ensure the agency is focused on the mission of protecting public health and safety and security while supporting increasing mandates.

The NRC is approaching work in the context of aligning the budget with changing priorities and strategically focusing on not only replacing employees who depart but also fine-tuning available skill sets to meet future mission needs. We are taking a fresh and realistic look at each of our business lines and corporate support lines and where we anticipate the agency's posture will be in five years so that we can adjust, refine, and redirect our activities as appropriate.

TECHNICAL SESSIONS
Tuesday, March 11, 2014, 1:30 p.m. – 3:00 p.m.

T1 Agency Efforts to Address the Cumulative Effects of Regulation

Session Chair: Lawrence Kokajko, Director, Division of Policy and Rulemaking, NRR/NRC, 301-415-1282, Lawrence.Kokajko@nrc.gov

Session Coordinator: Tara Inverso, Project Manager, Division of Policy and Rulemaking, NRR/NRC, 301-415-1024, Tara.Inverso@nrc.gov

Questions submitted during the above session were answered during the session's Q/A period.

T2 Interacting with the NRC

Session Chair: Mary Givvines, Deputy Office Director, OIS/NRC, 301-415-7330, Mary.Givvines@nrc.gov

Session Coordinator: Beth Deahl, Customer Relations Specialist, Customer Service Division, OIS/NRC, 301-415-5684, Elizabeth.Deahl@nrc.gov

The questions below were not answered during the above session.

QUESTION 1: When is it required to put internal NRC emails into ADAMS, for example between site inspectors and the region or Headquarters?

ANSWER 1 [answered by NRC]: Although emails generally are not required to be made publicly available absent a FOIA request that specifically seeks that record, the documents that are required for Proactive Disclosure are governed by 10 C.F.R. 9.21. These records are limited to six specific categories of information. One of those categories of information, governed by 10 C.F.R. § 9.21(c)(5), includes copies of records that have been released to a person under the FOIA that, because of the nature of their subject matter, are likely to become the subject of subsequent requests. As such, if a specific communication has been sough multiple times via FOIA, it will be reviewed for proactive disclosure. Additionally, responses to all other FOIA requests that can properly be made publicly available are accessible via ADAMS separate from NRC Proactive Disclosures. Many of these include these types of email communications between site inspectors, the regions, and Headquarters. Lastly, Program Office heads have the discretion, under 10 C.F.R. 9.25(f), to discretionarily release information (including emails) that are sought pursuant to a FOIA request and located within their Program Office during the search for records. This discretionary release can be made by the Program Office head so long as (1) the disclosure of the agency records will not be contrary to the public interest and (2) the disclosure will not affect the rights of any person.

<u>T3</u> **<u>International Regulators Q&A Panel Session</u>**

Session Chair: Jennifer Uhle, Deputy Director, Reactor Safety Programs, NRR/NRC, 301-415-1270, Jennifer.Uhle@nrc.gov

Session Coordinator: Gene Carpenter, Team Leader, Division of Inspection and Regional Support, 301-415-2983, Gene.Carpenter@nrc.gov

Questions that were not addressed during the session and their answers will be posted when they become available. In the meantime, if you have any questions pertaining to this topic, please contact the Session Chair or Coordinator listed above.

<u>T4</u> **<u>Is it Possible to Create a Risk-Informed and Performance-Based Emergency Preparedness Regulatory Regimen?</u>**

Session Chair: Robert Lewis, Director, Division of Emergency Preparedness and Response, NSIR/NRC, 301-287-3779, Robert.Lewis@nrc.gov

Session Coordinator: Carolyn Kahler, Emergency Preparedness Specialist, Division of Preparedness and Response, NSIR/NRC, 301-287-3722, Carolyn.Kahler@nrc.gov

Questions submitted during the above session were answered during the session's Q/A period.

<u>T5</u> **<u>Recent Operating Reactors Materials and Mechanical Component Issues</u>**

Session Chair: Robert Hardies, Senior Level Advisor for Materials Engineering, Division of Engineering, NRR/NRC, 301-415-5802, Robert.Hardies@nrc.gov

Session Coordinator: Kyle Hanley, Materials Engineer, Division of Engineering, NRR/NRC, 301-415-1975, Kyle.Hanley@nrc.gov

Questions submitted during the above session were answered during the session's Q/A period.

<u>T6</u> **<u>The Role of Technical Scientific Support Organizations within Nuclear Regulatory Authorities</u>**

Session Chair: Stuart Richards, Deputy Director, Division of Systems Analysis, RES/NRC, 301-251-7499, Stuart.Richards@nrc.gov

Session Coordinator: Don Algama, Reactor Systems Engineer, Division of Systems Analysis, RES/NRC, 301-251-7940, Don.Algama@nrc.gov

Questions submitted during the above session were answered during the session's Q/A period.

TECHNICAL SESSIONS
Tuesday, March 11, 2014, 3:30 p.m. – 5:00 p.m.

T7 International Challenges and Perspectives on Spent Fuel Storage

Session Chair: Mark Lombard, Director, Division of Spent Fuel Storage and Transportation, NMSS/NRC, 301-287-0673, Mark.Lombard@nrc.gov

Session Coordinator: Shawn Rochelle Smith, Senior International Program Coordinator, NMSS/NRC, 301-287-9184, Shawn.Smith@nrc.gov

The questions below were not answered during the above session.

QUESTION 1 [addressed to Darius Lukauskas, VATESI, Lithuania]: Your spent fuel storage facility is to have a hot cell for fuel examination. Is the hot cell going to be large enough to handle an entire dry storage cask?

ANSWER 1 [answered by Darius Lukauskas, VATESI, Lithuania]: As it is set up in the design of storage facility the primary purpose of hot cell is to inspect and repack spent nuclear fuel. Hence, dimensions of FIHC are enough to accept entire dry storage cask and temporary store the unloaded contents of the cask safely.

QUESTION 2 [addressed to Darius Lukauskas, VATESI, Lithuania]: Current Ignalina interim storage is outdoors - new one is indoors - why? Does the building have safety functions?

ANSWER 2 [answered by Darius Lukauskas, VATESI, Lithuania]: It was proposed such design of the storage facilities with other licensing submittals. Following buildings structures are considered as safety related for the new spent fuel storage facility: supporting structures of the storage building including foundation slab, reinforced concrete structures of hot cell, internal and external shielding walls. Roof of the storage building does not have safety functions.

QUESTION 3 [addressed to Darius Lukauskas, VATESI, Lithuania]: Are cask seals monitored by pressure gauges on both CASTOR and CONSTOR? If not, why not? Leak-tightness control is periodically carried out only for CASTOR casks. Leak-tightness is determined by taking a HeliTest probe in the casks shielding lid aperture and measuring of helium concentration in the inspected cavity.

ANSWER 3 [answered by Darius Lukauskas, VATESI, Lithuania]: Each CONSTOR cask is checked before transfer to the final storage (inventory correctness, drying criterion, tightness of the cask cavity, dose rate measurement, surface's temperature measurement, weld seams non-destructive checking). After positioning of cask in the storage facility, there are no significant operational loads, which could lead to cask damage. Casks shall be randomly inspected in the cask service station in new spent fuel storage facility. In the new storage facility also radiological monitoring of the exhaust air will be performed and a special program prepared to detect which cask is leaking.

QUESTION 4 [addressed to Darius Lukauskas, VATESI, Lithuania]: Any problems with CONSTOR transport?

ANSWER 4 [answered by Darius Lukauskas, VATES, Lithuania I]: CONSTOR RBMK1500 casks are dual-purpose casks for storage and transport of spent fuel. With an impact limiters CONSTOR RBMK1500 casks fulfill test criteria B(U)F type packages. CONSTOR RBMK1500/M2, which will be stored in the new spent fuel storage facility are also dual-purpose casks. Cask meets the transport regulations when fitted with the shock absorbers and overpack only. The shock absorbers and overpack are not part of the project for construction of new spent fuel storage facility as well as certification of the cask as B(U)F type package is not foreseen in the scope of the contract. However, it is reasonable to expect that the level of evidence of compliance with the IAEA transport regulations will enable approval against all relevant criteria.

It should be noted that the existing and the new spent fuel storage facilities are close to Ignalina NPP and transportation of casks from NPP to storage facilities by public rails is not needed. Transportation of casks is performed by rails owned by Ignalina NPP (internal rails). Both type of CONSTOR casks are considered by VATESI as suitable for storage and transportation to the storage facilities by internal rails. Such use of cask is justified in the Safety Analysis Reports.

QUESTION 5 [addressed to Darius Lukauskas, VATESI, Lithuania]: What is the source of funding for the interim storage or the permanent repository, in terms of licensing, construction, transportation, etc.?

ANSWER 5 [answered by Darius Lukauskas, VATESI, Lithuania]: There are several financing sources for management of radioactive waste and spent fuel in Lithuania: national State Enterprise INPP Decommissioning Fund, State budget, Ignalina International Decommissioning Support Fund, Ignalina Programme. These new facilities, such as spent nuclear fuel storage facility are being financed by the Ignalina International Decommissioning Support Fund, Ignalina Programme and co-financed by the national State Enterprise INPP Decommissioning Fund or State budget.

The Ignalina International Decommissioning Support contains contributions of the country-donors, where the main contributor is European Commission. The European Bank for Reconstruction and Development is the administrator of the Ignalina International Decommissioning Support Fund, while the governing body is the Donors Assembly.

The Ignalina Programme is financed by the European Union budget. The Ignalina Programme was created under Protocol 4 of the Act of Accession of Lithuania into the European Union in order to provide assistance for the decommissioning of INPP (including radioactive waste management) and consequential measures in the energy sector. Programme is administrated by national Central Project Management Agency.

Decision on construction of deep geological repository in Lithuania is not taken yet. Hence, financing scheme for this facility construction not established.

QUESTION 6 [addressed to Darius Lukauskas, VATESI, Lithuania]: What is your policy with regards to storing spent fuel in dense packed arrangements in spent fuel pools? Do you require fuel to be moved after a certain period in the pool?

ANSWER 6 [answered by Darius Lukauskas, VATESI, Lithuania]: VATESI has not prescriptive technical requirements regarding the storage or movement of spent fuel in the storage pools. Hence, there are no requirements regarding the particular storage period of spent fuel in the cooling pool and etc. Ignalina NPP has prime responsibility for safety and has to demonstrate it to VATESI. Safety assessment reports and operational manuals of the NPP Units are subject for VATESI approval. Operational limits and conditions are set out in the operational manual. Licence holder shall develop operational procedures and instructions. These documents establish detailed conditions and requirements, which shall be met by licence holder. Hence, configuration of the spent fuel assemblies storage in the pool (storage grids), minimum storage period of spent fuel assemblies in the fuel pool before separation of fuel bundles from the assemblies and minimum storage period in the pool including storage of fuel bundles loaded into the baskets are established in the Ignalina NPP documents based on safety justification, design of Ignalina NPP and instructions of producer of nuclear fuel.

QUESTION 7 [addressed to Darius Lukauskas, VATESI, Lithuania]: For more than 50 years we are promising disposal of spent fuel, but we have disposed none so far. Do we need disposal at all? Why don't we simply admit to store it as long as some other use of that material will be found? Nobody could know today what technologies will exist in 50-100 years!

ANSWER 7 [answered by Darius Lukauskas, VATESI, Lithuania]: Depending on the nuclear fuel cycle policy spent fuel can be regarded either as a valuable resource that may be reprocessed or as radioactive waste that is destined for direct disposal. In both cases disposal of high level waste, separated at reprocessing, or of spent fuel regarded as waste cannot be avoided. Strategy for management of spent fuel and radioactive waste shall cover all stages from radioactive waste generation to disposal. Taking into account current knowledge it can be concluded that deep geological disposal is the safest and most sustainable option as the final stage of the management of high-level waste and spent fuel. Although technologies of radioactive waste management are evolving, their development cannot be considered as rapid. It is difficult to predict when other use of material, which could be a real alternative for disposal could occur. Hence, the storage of radioactive waste, including long-term storage, is just an interim solution. However, strategy for management of spent fuel and radioactive waste shall include a final solution. For other options than disposal it seems there is too much uncertainty, when and what the alternative final solution could be.

QUESTION 8 [addressed to Darius Lukauskas, VATESI, Lithuania]: Is Lithuania considering selling its fuel to be recycled or buying recycling of its spent fuel? If not, how long does Lithuania believe it can store the fuel in dry storage?

ANSWER 8 [answered by Darius Lukauskas, VATES, Lithuania I]: According to current Radioactive Waste Management Strategy (approved in 2008) interim storage is the chosen option of spent fuel and long lived radioactive waste management. One of the tasks formulated in the Strategy is analyze possibilities to dispose spent fuel and long-lived radioactive waste in Lithuania or to reprocess or dispose it in other countries. Hence, reprocessing is one of the possible options for spent fuel management after interim storage. However, it is not urgent issue to take decision and it could be considered later. Up to date RMBK spent fuel was not reprocessed anywhere due to its low enrichment. Design lifetime of the interim storage facilities is 50 years. Extension of the storage period could also be possible, if adequate safety justification provided.

QUESTION 9 [addressed to Darius Lukauskas, VATESI, Lithuania]: Size of a cask seems lower than size of an assembly. How is it managed?

ANSWER 9 [answered by Darius Lukauskas, VATESI, Lithuania]: Fuel assembly that total length is ~ 10 m consist of extension rod and two fuel bundles. Fuel assemblies are unloaded of from the reactor and placed for cooling into cooling pools (cooling period at least 1 year). After that period extension rods are removed, the rest of fuel assemblies are cut into two parts in a “hot cell” at NPP units. Fuel bundles, which length is ~ 3.6 m are placed in the baskets and these baskets are placed into casks for dry storage. Casks were designed to be suitable for baskets. Height of the casks is more than 4 m.

QUESTION 10 [addressed to Kirsi Alm-Lytz, STUK, Finland]: What is the source of funding for the interim storage or the permanent repository, in terms of licensing, construction, transportation, etc.?

ANSWER 10 [answered by Anna Lahkola & Kirsi Alm-Lytz, STUK, Finland]: The source of funding for all of the above mentioned operations is taken care of the licensees i.e. the nuclear operators that are private companies in Finland. Besides this, the state manages a secured waste management fund for circumstances where the nuclear operator is not able to take care of its waste management or disposal. The nuclear operators are also required to cover (pay fees to) the secured fund. As stated in the Nuclear Energy Act, after the waste management and disposal is taken care of safely and the disposal facility is closed, the responsibility of the spent nuclear fuel, as well as of the repository itself, will be then transferred to the state. At the time of the transfer, the nuclear operator is required to pay a lump sum of estimated future costs that comprise of continued institutional control or monitoring etc.

QUESTION 11 [addressed to Kirsi Alm-Lytz, STUK, Finland]: What is your policy with regards to storing spent fuel in dense packed arrangements in spent fuel pools? Do you require fuel to be moved after a certain period in the pool?

ANSWER 11 [answered by Anna Lahkola & Kirsi Alm-Lytz, STUK, Finland]: In the Finnish wet storages, there are both traditional and dense storage arrangements (wide and tight/dense racks). Both are designed, presented in the application to the regulator and approved by the regulator for permanent purposes. So, there are no requirements that the fuel is to be transferred from their positions in certain period, or ever, for that cause (i.e. only for final disposal or for other reason such as evacuation of the particular pool for some other reason).

QUESTION 12 [addressed to Kirsi Alm-Lytz, STUK, Finland]: For more than 50 years we are promising disposal of spent fuel, but we have disposed none so far. Do we need disposal at all? Why don't we simply admit to store it as long as some other use of that material will be found? Nobody could know today what technologies will exist in 50-100 years!

ANSWER 12 [answered by Anna Lahkola & Kirsi Alm-Lytz, STUK, Finland]: As a regulator we do not take opinions that may be political in nature. In Finland, the official policy of the state is direct disposal and as a regulator we are responsible for ensuring that it is safe. Therefore we can't speculate on the matter and really provide any answer to the questions.

QUESTION 13 [addressed to Holger Völzke, BAM, Germany]: What is the justification that the seal pressure change will remain linear to long time frames when testing included only 48 months?

ANSWER 13 [answered by Holger Völzke, BAM, Germany]: At this time linear seal pressure decrease (over logarithmic time scale) is identified in all tests consistently and it is a justified prediction to continue like this which of course has to be verified by continuing the tests for significantly longer periods of time.

QUESTION 14 [addressed to Holger Völzke, BAM, Germany]: Why is irradiation degradation considered low importance for the (U)HMW-PE, considering the known degradation modes of polymers in irradiated environments?

ANSWER 14 [answered by Holger Völzke, BAM, Germany]: Investigations so far have shown that degradation modes and microstructural changes do not lower the material's neutron shielding ability significantly. In parallel, neutron doses decrease more significant during interim storage periods. However, this expectation has to be verified by further investigations.

QUESTION 15 [addressed to Holger Völzke, BAM, Germany]: How many fuel assemblies are stored in your dual purpose casks?

ANSWER 15 [answered by Holger Völzke, BAM, Germany]: The latest cask generation of the Castor V – type e. g. contains 19 PWR or 52 BWR fuel assemblies.

QUESTION 16 [addressed to Holger Völzke, BAM, Germany]: Some people in the U.S. advocate storing fuel in small packages (4 assemblies or less) that might also be amenable to direct disposal. What do you think of that idea?

ANSWER 16 [answered by Holger Völzke, BAM, Germany]: Generally smaller disposal packages with lower mass and heat dissipation might be easier to handle and store in a deep geological repository. However, also bigger casks like the current interim storage packages have to be considered for direct disposal as a probably technically feasible option. So in both cases direct final disposal of interim storage packages might be a future option and it cannot be decided yet which option lastly might be the better one.

QUESTION 17 [addressed to Holger Völzke, BAM, Germany]: What is easier to provide extended storage for: used nuclear fuel or high level waste from recycling/reprocessing?

ANSWER 17 [answered by Holger Völzke, BAM, Germany]: With regard to the casks used in Germany there is no difference. Concerning the contents fuel assemblies are more fragile than robust stainless steel canisters filled with the stable glass matrix. Significant differences exist with regard to potential activity releases. In case retrievability of the contents after extended interim storage is required fuel assemblies are more challenging than vitrified HLW.

QUESTION 18 [addressed to Holger Völzke, BAM, Germany]: What are the consequences of seal creep? Leakage under normal conditions? Leakage under accident conditions? If accident, does BAM consider a brief “burp” acceptable?

ANSWER 18 [answered by Holger Völzke, BAM, Germany]: Generally, seal creep results in lower pressure forces and useable elastic resilience depending on time and temperature. At the same time seal effect increases due to tighter contact between seal surfaces. Increased leakage under normal conditions of transport should not be an issue. In case of accident scenarios this has to be evaluated in the future by investigating aged systems. Basically, whether a “burp” might be acceptable or not depends on potential activity release from such an event.

QUESTION 19 [addressed to Holger Völzke, BAM, Germany]: What is the source of funding for the interim storage or the permanent repository, in terms of licensing, construction, transportation, etc.?

ANSWER 19 [answered by Holger Völzke, BAM, Germany]: Interim storage is funded directly by the waste producer whether private or state-owned. This includes construction, licensing, operation, etc. Final disposal is organised and funded by the federal government based on a continuously prepaid fond (paid by all waste producers as well).

QUESTION 20 [addressed to Holger Völzke, BAM, Germany]: What is your policy with regards to storing spent fuel in dense packed arrangements in spent fuel pools? Do you require fuel to be moved after a certain period in the pool?

ANSWER 20 [answered by Holger Völzke, BAM, Germany]: With the German face-out decision all fuel from the shut-down NPP’s will be removed from the sites and put into dry interim storage as soon as possible.

QUESTION 21 [addressed to Holger Völzke, BAM, Germany]: For more than 50 years we are promising disposal of spent fuel, but we have disposed none so far. Do we need disposal at all? Why don’t we simply admit to store it as long as some other use of that material will be found? Nobody could know today what technologies will exist in 50-100 years!

ANSWER 21 [answered by Holger Völzke, BAM, Germany]: Principally, this statement is not in congruence with today’s waste management policies which require a final solution to be realized by the generation who used nuclear fuel for their own benefit. On the other hand we all know that in many countries the establishment of a final repository will probably take at least some more decades. That means interim storage will be needed for many decades, may be more than 100 years. In that case technical, scientific and socio-economic developments will, of course, affect future decisions, especially if new technologies may become available. But from today's perspective deep geological disposal is the only globally accepted solution for the safe and secure final disposal of spent nuclear fuel.

QUESTION 22 [addressed to Elena Mantagaris, NWMO, Canada]: What is the source of funding for the interim storage or the permanent repository, in terms of licensing, construction, transportation, etc.?

ANSWER 22 [answered by Elena Mantagaris, NWMO, Canada]: The Government of Canada selected Canada's plan for the long-term management of used nuclear fuel in June 2007. The approach, called Adaptive Phased Management (APM), calls for the construction of a deep geological repository to safely and securely contain and isolate Canada's used nuclear fuel, and includes a Centre of Expertise for technical, environmental and community studies.

The planning, development and implementation of the project is funded by the major owners of used nuclear fuel in Canada: Ontario Power Generation (OPG), New Brunswick Power, Hydro-Québec and Atomic Energy of Canada Limited. The Nuclear Fuel Waste Act (NFWA) requires each of these four companies to establish independently managed trust funds and make annual deposits to ensure that the money to fund this project will be available when needed. Contributions by each waste owner are determined according to a funding formula and based on the amount of waste generated. The funds were established in 2002, and annual contributions have been made by each waste owner since. Legislation provides that the NWMO may only access the trust funds for implementing the management program once a construction or operating licence has been issued by the regulator.

We encourage you to read more about this topic in our 2014 backgrounder Financial Surety and Updated Lifecycle Cost Estimate for Adaptive Phased Management, which can be found on our web site at http://www.nwmo.ca/backgrounders.

QUESTION 23 [addressed to Elena Mantagaris, NWMO, Canada]: What is your policy with regards to storing spent fuel in dense packed arrangements in spent fuel pools? Do you require fuel to be moved after a certain period in the pool?

ANSWER 23 [answered by Elena Mantagaris, NWMO, Canada]: Used nuclear fuel is currently managed at interim storage facilities by waste owners, such as Ontario Power Generation (OPG). We took the liberty of sharing this question with OPG. Here is the response they provided:

As spent fuel bundles are removed from the reactor new fuel is loaded. After removal from the reactors, by using remote controlled equipment, the used fuel is stored in water pools, called fuel bays, located in the stations. The bundles are placed in modules and carefully managed within the fuel bay. To ensure storage space is available to accept used fuel bundles the bundles remain in the fuel bays for about 10 years to cool and then they are loaded into dry storage containers and taken to the dry storage facility. OPG moves the fuel out of the wet bays to make room for more fuel.

The decision was made decades ago not to expand the fuel bays to house the entire inventory of spent fuel from the station's production. Instead OPG designed transportable dry storage containers and above ground dry storage buildings to store them, and ready for transport to a final long-term storage facility.

The used fuel bay looks like a swimming pool, but is built of reinforced concrete with a stainless steel liner and is designed to withstand earthquakes and detect leaks. The water in the Used Fuel Bays is demineralised water that is constantly processing through a closed loop filtration system of Ion Exchange columns and heat exchangers to remove any particulate and residual heat. The water in the used fuel bays provides cooling and shielding for the fuel bundles. Each fuel bundle, at removal from the reactor, generates about six kilowatts of heat which is equal to turning on a kitchen stove with all 4 burners on 'high'. One year later it radiates the same amount of heat as a 60 watt light bulb and one percent of the radiation is left.

QUESTION 24 [addressed to Elena Mantagaris, NWMO, Canada]: For more than 50 years we are promising disposal of spent fuel, but we have disposed none so far. Do we need disposal at all? Why don't we simply admit to store it as long as some other use of that material will be found? Nobody could know today what technologies will exist in 50-100 years!

ANSWER 24 [answered by Elena Mantagaris, NWMO, Canada]: APM, and the repository planned as its endpoint, meets the priority of Canadians to take action now to provide a long-term management approach for the used fuel currently stored in Canada on an interim basis at the reactor sites where it is generated.

Although its radioactivity decreases with time, chemical toxicity persists and the used fuel will remain a potential hazard for many hundreds of thousands of years. For this reason, used fuel requires careful management essentially indefinitely. Canada's used fuel is now safely stored on an interim basis at 7 licensed surface facilities located at the nuclear power plants where it is produced. However, this interim storage requires ongoing care over the very long time frame for which used nuclear fuel needs to be managed.

The Adaptive Phased Management (APM) plan for the safe, long-term management of Canada's used nuclear fuel requires that it be centralized in a single location and contained and isolated in a deep geological repository in a suitable rock formation. This approach emerged from a three-year study of options with Canadians and reflects their preferences and priorities. Led by the NWMO, the study engaged Canadians in every province and territory on the issue. Canadians have emphasized that safety and security are the top priority now and in the future, and that this generation must assume active responsibility for putting in place a plan for the long-term stewardship of used nuclear fuel. Ensuring the long-term, safe and secure management of used nuclear fuel for the protection of people and the environment is an important responsibility we all share.

QUESTION 25 [addressed to Elena Mantagaris, NWMO, Canada]: With so much unoccupied land in Canada, why has NWMO considered population centers for the long-term repository? Why not less populated areas, even if fewer people reap economic benefits?

ANSWER 25 [answered by Elena Mantagaris, NWMO, Canada]: The process for identifying an informed and willing host community for a deep geological repository is designed to ensure, above all, that the site selected is safe and secure for people and the environment, now and in the future. Detailed field investigations involving geophysical surveys, characterization of the existing environment, drilling and sampling of boreholes, field and laboratory testing, and monitoring activities will be conducted during site characterization to affirm the suitability of the site. The NWMO will be required to demonstrate the safety of the repository and its components to the satisfaction of the Canadian Nuclear Safety Commission in order to receive a license to construct the facility.

It is important to note that the used nuclear fuel has been safely stored in large urban centers for more than 40 years. Low population density is not a criterion for selection of a safe site.

The NWMO is seeking an informed and willing host. Only communities that are interested in the project, and that have expressed this interest, will be considered. The project will not be imposed on any community.

Currently, 15 communities are engaged in learning about Canada's plan and the siting process. The siting process involves a number of steps. A community will proceed from one step to the next only if it chooses to do so and if the work to assess the suitability of the site supports it. Over time and through increasingly detailed studies it will become clearer which communities have the strongest potential to safely host the project. The process for identifying an informed and willing host for a deep geological repository is designed to ensure, above all, that the site selected is safe and secure for people and the environment, now and in the future.

Ultimately, the project will only proceed with the involvement of the interested community, Aboriginal communities and other surrounding communities working together to implement it.

T8 Licensee Supply Chain Challenges with Vendors and Subtier Suppliers

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The questions below were not answered during the above session.

QUESTION 1: How specific are supply chains to specific plant constructions projects?

ANSWER 1 [answered by Richard Rasmussen, NRC]: Supply chains are determined by the purchaser and are a matter of free enterprise. Therefore, they are likely to vary by project.

ANSWER 1 [answered by Steve Napiecek, NNI]: This question needs to be addressed to the WEC representative.

ANSWER 1 [answered by Peter Varga, WEC]: Supply chains are not project specific for the most part. However, we do have commitments to host countries and regions to buy where we build. This commitment is not all inclusive and is limited to services and commodities that the host region can accommodate in a qualified and capable fashion.

ANSWER 1 [answered by Sungho Yang, KINS]: The supply chains of safety class 1 components are relatively simple since there are not many manufacturers who can supply such materials or components. But for safety class 2 and 3 components such as valves, pumps, or I&C items that are commonly used in the non-nuclear industry, supply chains are depending on the country's industry infrastructure. Korea has relatively well-developed heavy industry infrastructure.

QUESTION 2: Would standardizing supply chains improve the safety performance of nuclear projects?

ANSWER 2 [answered by Richard Rasmussen, NRC]: Perhaps, but NRC is not in a position to regulate the supply chain. That responsibility rests with the licensees.

ANSWER 2 [answered by Peter Varga, WEC]: Standardizing supply chains, assuming you mean that we consistently use only the most reliable vendors of nuclear safety related products and services, again and again, would increase the level of assurance of first time quality. First time quality equates to defect free plants and therefore high reliability in the area of nuclear safety performance. This mentality applies to industrial safety performance at the construction sites as well. Industrial safety posture comes from refined and high standards, plus a proven track record of implementing those standards across the organization.

ANSWER 2 [answered by Sungho Yang, KINS]: Yes. It will be helpful. And I would recommend the development of a reliable material organization.

QUESTION 3: Is an international standardization possible?

ANSWER 3 [answered by Richard Rasmussen, NRC]: Standardization of codes and standards is a topic of continued interest to regulators and vendors. Although some improvement has been made, full standardization would likely require changes to laws imposed by individual countries.

ANSWER 3 [answered by Peter Varga, WEC]: Yes. Westinghouse's procurement strategy is already very much international.

ANSWER 3 [answered by Sungho Yang, KINS]: Since Appendix B QA program is applicable for the vendors other than pressure component manufacturers, ASME NQA-1 Certificate would be one way of such international standardization.

QUESTION 4: The supply chain is tightly controlled. When/Where do innovations occur?

ANSWER 4 [answered by Richard Rasmussen, NRC]: The NRC does not consider the quality requirements of Appendix B to 10 CFR 50 to be contrary to innovation.

ANSWER 4 [answered by Steve Napiecek, NNI]: This question needs to be addressed to the WEC representative.

ANSWER 4 [answered by Peter Varga, WEC]: Innovations can occur everywhere and at any time, prompted in part by robust Corrective Action Programs and Lessons Learned Programs. Innovation occurs in Logistics, Design, Engineering… etc. The AP1000 plant design itself is an innovative (passive) design. The SMR design efforts are innovated. Many of the AP1000 components are the result of innovation (VFD's, squib valves). Logistics folks are always looking for better, faster, more reliable methods to package and ship from supplier to end user.

ANSWER 4 [answered by Sungho Yang, KINS]: I think tight control of the supply chain does not interfere with innovation.

QUESTION 5: How do you define "safety culture?"

ANSWER 5 [answered by Richard Rasmussen, NRC]: The NRC Safety Culture Policy Statement defines nuclear safety culture as the core values and behaviors resulting from a collective commitment by leaders and individuals to emphasize safety over competing goals to ensure protection of people and the environment. Links to the NRC safety culture policy statement can be found on the NRC home page.

ANSWER 5 [answered by Steve Napiecek, NNI]: One of the more text book answers is the organizations behaviors resulting from a collective commitment by leaders and individuals to emphasize safety over competing goals to ensure protection of people and the environment. At a module fabrication company, we stress the following behaviors and actions: 1) Promotion of a Safety Conscious Work Environment where are employees are encouraged to raise any safety concern which are appropriately resolved with timely feedback to the originator of the concern; 2) Leadership Safety Values and Behavior where senior management are on the production floor every shift; 3) Problem Identification and Resolution; and 4) a Questioning Attitude to continuously challenge existing conditions and activities with employees empowered to STOP any activity when questions of safety or procedure compliance are raised.

ANSWER 5 [answered by Peter Varga, WEC]: INPO and the NRC define safety culture as the core values and behaviors resulting from a collective commitment by leaders and individuals to emphasize safety over competing goals to ensure protection of people and environment. An important part of a healthy safety culture is a safety-conscious work environment (SCWE). SCWE is an environment in which personnel feel free to raise safety concerns without fear of retaliation, intimidation, harassment or discrimination, where concerns are promptly reviewed, given the proper priority based on their potential safety significance, and appropriately resolved with timely feedback. Westinghouse subscribes to this definition.

ANSWER 5 [answered by Sungho Yang, KINS]: The safety of nuclear facilities can be secured through dedication to common goals for nuclear safety by organizations and individuals at all levels by giving a high priority to safety through sound thoughtful knowledge and a proper sense of safety responsibility. The Korean Regulator recognizes that nuclear safety is achieved not only by safety systems and strict regulations but also by the spread of safety culture. In meeting this commitment, the Regulator strives for strict regulations through the development of clear safety goals and regulatory policies. It will actively encourage safety-related research and technical development to achieve technical expertise of regulatory activities and will ensure regulatory independence and fairness by minimizing any undue pressure and interference. Nuclear utilities establish management policies, giving a high priority to nuclear safety, and foster a working climate in which attention to safety is a matter of everyday concern. Managers encourage, praise and provide tangible rewards to employees for commendable attitudes and good practices concerning safety matters. On the contrary, when errors are committed, individuals are encouraged to report them without any concealment and to correct them to avert future problems. For repeated deficiencies in or negligent attitudes toward nuclear safety, managers take firm measures in such a way to prevent the same errors from occurring again. In this way, safety culture will be achieved through sound safety policies and full understanding of safety culture by the senior management and through proper practices and implementation by individuals engaged in the nuclear industry.

QUESTION 6: The speakers focused on new plan construction but supplier issues greatly affect operating plants as well. Any comments on operating plants? Why not include in this panel?

ANSWER 6 [answered by Richard Rasmussen, NRC]: The examples provided in this session applied to both new and operating reactor sites.

ANSWER 6 [answered by Steve Napiecek, NNI]: This question needs to be addressed either the utility representative or the WEC representative.

ANSWER 6 [answered by Peter Varga, WEC]: Supplier quality and supplier quality oversight for procurement activities for operating plants is equally challenging as procurement for new construction plants.

QUESTION 7: We have heard about nuclear new build supply chain issues. From the utility perspective, how are you managing the equipment adolescence and life extension issues at operating plans, especially since most vendors do not have nuclear QA programs.

ANSWER 7 [answered by Peter Varga, WEC]: I cannot provide an answer from the utility perspective since I don't work for a utility. However, Westinghouse is in the business of providing nuclear services to the operating fleet, including full lifecycle support, cradle to grave. The utilities can make a wise decision by turning to Westinghouse for help in managing equipment adolescence and life extension issues. We have a several organizations that specialize in this type of service.

ANSWER 7 [answered by Sungho Yang, KINS]: In Korea, Korea Electric Power Industry Code is maintained by the Korea Electric Association (KEA), and about 250 vendors are certified by the KEA's Nuclear Accreditation Program. The KEA certifies not only pressure boundary component vendors but also electrical component vendors. Even in this situation Korea Hydro and Nuclear Power Company (KHNP), the unique utility in Korea, also faces the life extension issue and to resolve nuclear item procurement issues KHNP participates in NUPIC and also established their own dedication organization.

QUESTION 8 [addressed to Peter Varga, WEC]: What practice do you use to incentivize Tier 1 suppliers to have a robust sub-supplier oversight program?

ANSWER 8 [answered by Peter Varga, WEC]: Our Contracts personnel attempt to motivate supplier behavior and performance by employing contract language, and Terms and Conditions that clearly define responsibilities, accountabilities and consequences. It is not an exact science and the results are not exactly repeatable and reproducible, but good contract language can be effective when coupled with robust oversight.

QUESTION 9 [addressed to Mark Rauckhorst, Southern Nuclear and Peter Varga, WEC]: Please comment on a "low price wins" mentality versus paying for the "value" brought by a mature nuclear fabricator. That is, what is the "value" worth?

ANSWER 9 [answered by Peter Varga, WEC]: A strategy that is based on "low price wins" is a bankrupt strategy. The "low price" strategy (to me) means the low bid wins without proper consideration or credence given to forecasted errors, first time quality, COPQ, potential violations, rework, repairs, etc. It says nothing about the elevated risk of defects going unnoticed during fabrication and ending up with the Customer. The "low price wins" strategy is a risky strategy, where the end user is unknowingly assuming a risk passed on by the supplier. The "low price wins" strategy has no place in the nuclear industry.

QUESTION 10 [addressed to Peter Varga, WEC]: What are the resource demands on licensees to ensure supplier oversight down the supply chain?

ANSWER 10 [answered by Peter Varga, WEC]: Westinghouse is not a licensee so I'm afraid I cannot answer this question. However, Westinghouse has invested heavily in a large quality oversight task force that is positioned all over the globe. Our most recent challenges have been occurring at Tier 2 and lower tier suppliers. So we've had to invest even more in resources, and we've had to negotiate access rights.

T9 Medical Radioisotope Production: U.S. Efforts to Establish a Reliable Domestic Supply of Molybdenum-99

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The questions below were not answered during the above session.

QUESTION 1 [addressed to Steve Lynch, NRC]: Is there any current consideration for using a Part 52 type licensing approach for medical isotope production facilities?

ANSWER 1 [answered by Steve Lynch, NRC]: 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," governs the issuance of early site permits, standard design certifications, combined licenses, standard design approvals, and manufacturing licenses for nuclear power facilities. As such, facilities other than nuclear power plants, such as medical isotope production facilities, fall outside of the scope of this licensing approach.

QUESTION 2 [addressed to Steve Lynch, NRC]: With the changes to include accelerator based byproduct materials under Part 30 materials has there been consideration of something separate from Part 50 to apply to accelerators construction and operation licensing, since Part 50 is very reactor centric?

ANSWER 2 [answered by Steve Lynch, NRC]: The NRC has not considered applying 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material," regulations to the construction and operation licensing of accelerators. While licensees must meet the applicable regulations contained in 10 CFR Part 30 that govern the domestic licensing of byproduct material, the issuance of construction permits and operating licenses to proposed medical isotope production facilities falls outside of the scope of this part of the regulations.

QUESTION 3 [addressed to Steve Lynch, NRC]: Will a facility such as SHINE be subject to 10 CFR Part 74 that requires accounting of their nuclear materials? If so, how would SHINE handle the constantly changing amount of material?

ANSWER 3 [answered by Steve Lynch, NRC]: Yes, SHINE will be subject to all applicable regulations contained within 10 CFR Part 74, "Material Control and Accounting of Special Nuclear Material," and SHINE's compliance with these regulations will be verified through the NRC's oversight program. Additionally, SHINE must meet applicable requirements of 10 CFR

Part 70, “Domestic Licensing of Special Nuclear Material,” to receive title to, own, acquire, deliver, receive, possess, use, and transfer its proposed amounts of special nuclear material.

QUESTION 4 [addressed to Steve Lynch, NRC]: How many expressions of interest has NRC received for possible new medical isotope production facilities?

ANSWER 4 [answered by Steve Lynch, NRC]: As of March 1, 2014, the U.S. Nuclear Regulatory Commission has received eight letters of intent proposing the construction and operation of medical isotope production facilities.

QUESTION 5 [addressed to Steve Lynch, NRC]: [Are there or will there be] regional inspectors of medical isotope reactors?

ANSWER 5 [answered by Steve Lynch, NRC]: NRC staff will inspect medical isotope production facilities to ensure that the licensees safely conduct NRC-regulated activities.

QUESTION 6 [addressed to Orhan Suleiman, FDA or Parrish Staples, DOE]: Will the lack of a strong reliable HEU free supply or Tc99m push more use of alternate isotopes? If yes, what are the dose implications to patients?

ANSWER 6 [answered by Orhan Suleiman, FDA]: No. I feel there are enough alternative ways to produce Mo99, either with LEU, or accelerators, that if there is a need, Mo99/Tc99m will always be produced and available. However, if shortages actually occur that scheduling cannot resolve, then the medical community may consider alternative procedures. I do not consider this to be a high probability. There are many global reactors who want to produce Mo99 that currently do not.

Alternatives tests vary in many ways from Tc99m. Current alternatives are thalium-201 or Rubidium-82 cardiac scans, fluoroscopic angiography, computed tomography angiography, ultrasound, or magnetic resonance angiography. Not only do each of these alternative tests provide different types of information, they each give the patient very different radiation absorbed doses.

For cardiac imaging, thallium-201 will give a higher dose (2x’s) than Tc99m, whereas Rb-82 will give a lower dose (1/10). Fluoroscopic angiography will probably give a much higher dose, but the dose can vary by an order of magnitude. CT Angiography may be comparable to Tc99m, but x-ray based technologies can easily give doses that vary by an order of magnitude; so comparisons are difficult. Then there are the nonionizing radiations, such as MRI and ultrasound, which give zero dose to the patient.

Ultimately, in medicine, it will depend on how ill the patient is, and what the value, or the efficacy of the test will be. If Tc99m tests provide equivalent or superior diagnostic information, they will remain viable! This is the ultimate question as evidence-based medicine evolves, if the test provides useful information in terms of patient management and patient outcome, it will be the predictor of whether or not that test will be used. Eventually, which alternative will be used will depend on availability of the Tc99m, and whether the patient can be rescheduled; and if not, which alternative test will the physician want performed. Radiation dose and cost will probably be secondary considerations.

ANSWER 6 [answered by Parrish Staples, DOE]: The shortages experienced over the past few years are evidence of a fragile supply chain. While this fragility has seen some movement to alternative isotopes in certain instances, Mo-99/Tc-99m remains the most widely used isotope for diagnostic imaging. As the Mo-99 community transitions to full cost recovery, non-HEU-based production, and improved reliability of supply is established, if the demand for the medical community continues, it is likely that Mo-99/Tc-99m will continue to be the industry standard.

QUESTION 7 [addressed to Orhan Suleiman, FDA]: With PET imaging becoming more prevalent, has the thought pattern been addressed to focus more on making cyclotrons more affordable instead of a Moly source?

ANSWER 7 [answered by Orhan Suleiman, FDA]: Yes, the "thought pattern" may be considering using more cyclotrons, but I do not see such production will truly threaten conventional Mo99 production. Accelerator produced Mo99 will play a role, but I do not see it eliminating the reactor production of Mo99. Similar to my last answer, it will depend. PET agents are more difficult to prepare, and require a PET scanner. It is possible if accelerator production becomes widespread, it will make the access and cost of PET agents competitive. However, Tc99m is widely used because it is easily produced, widely accepted, and inexpensive. This is why it is used in 30 million exams globally, and represents 80% of all nuclear medicine exams. I believe the medical demand for the product will drive the technology. Tc99m is an ideal radionuclide, and can be labeled with many drugs. PET drugs are more expensive, and unless they are really superior to alternative tests, will have a steeper hill to climb. Tc99m is not going away!

QUESTION 8 [addressed to John Adams, NRC]: What is the role of PRA for reviewing applications? What risk frequency criteria will be used?

ANSWER 8 [answered by John Adams, NRC]: Non-power reactors (NPRs) in the United States have been licensed using 10 CFR Part 50. Implementation of 10 CFR Part 50, as it applies to NPRs, has been achieved using only deterministic methods and acceptance criteria. Licensing decisions allowing construction and operation of NPRs have focused on assurance that worker and public doses are maintained within the limits contained in 10 CFR Part 20 for research reactors and 10 CFR Part 100, "Reactor Site Criteria," for test reactors. As was the case with power reactors, a set of licensing-basis events was established that was intended to ensure conservatism in design and protection from a wide spectrum of postulated events, up to and including design-basis accidents (DBAs). Those accidents are highly stylized and do not consider multiple failures of safety systems. Qualitative approaches for ensuring reliable safety systems, such as the single failure criterion, were implemented. Testing plans and operational limits were established in technical specifications to ensure that if called upon safety systems would perform. Unique to the licensing of NPRs includes the analysis of a maximum hypothetical accident (MHA). Analysis of the MHA is necessary because many NPRs are designed and operated so that an accident involving a radioactive release is not credible. The MHA assumes an incredible failure that results in consequences, including an (incredible) radioactive release, that bound all credible DBA consequences. Because the MHA is not expected to occur, only the potential consequences are analyzed and not the initiating event and scenario details.

The assessment of risk at NPRs has been qualitative and based on conservative deterministic assumptions, traditional engineering analyses, and operational experience. To date, the operators of NRC-licensed NPRs have not used modern risk assessment methods, such as probabilistic risk assessment methods, in support of licensing activities. No reactor safety goals and objectives that parallel the Commission's Safety Goal Policy Statement (NRC, 1986) have been developed for NPRs. Recently, Commissioner Apostolakis' Risk Management Task Force developed NUREG-2150, "A Proposed Risk Management Regulatory Framework," (Agency Document Access Management System (ADAMS) Accession No. ML12109A277) that includes a recommendation associated with NPRs to evaluate the utility of performing a pilot risk assessment, including consideration of external hazards, using modern risk assessment methods at an NPR. This evaluation, if conducted, would assess the value of the risk insights gained from the risk assessment on the basis of possible safety enhancements and possible contributions to a more efficient and effective risk-informed and performance-based regulatory framework for NPRs. Consideration of this and other recommendations included in NUREG-2150 are currently underway by NRC staff and have been the subject of two public meetings and publication of an early draft of a proposed policy statement (Pages 70354 – 70356 [FR DOC # 2013–28065]). Additional public meetings and opportunities for public comment are anticipated as this work progresses.

QUESTION 9 [addressed to Greg Piefer, SHINE]: How much H-3 gas is involved in the accelerator system and how well will it be confined?

ANSWER 9 [answered by Greg Piefer, SHINE]: The amount of tritium gas involved in the SHINE process is Security-Related, and therefore withheld from public disclosure. The tritium is contained within process systems, gloveboxes, and confinement boundaries within the facility to ensure it is properly contained. SHINE will ensure releases of tritium to the facility or environment are within 10 CFR 20 limits and maintained as low as reasonably achievable.

QUESTION 10 [addressed to Greg Piefer, SHINE]: What is the source for tritium? Do you anticipate any issues maintaining an adequate supply?

ANSWER 10 [answered by Greg Piefer, SHINE]: SHINE has identified multiple sources for tritium, and does not anticipate issues maintaining an adequate supply.

QUESTION 11 [addressed to Greg Piefer, SHINE]: What are the waste processing or isotope separation concerns for the other products in the Uranium solution? What are the anticipated contamination control issues of using a liquid solution target?

ANSWER 11 [answered by Greg Piefer, SHINE]: The fission process will generate a range of isotopes in addition to Mo-99, including some other medically-important isotopes that SHINE is planning on providing to the market. SHINE is developing separation and purification processes that specifically isolate the isotopes of interest given any of the potential impurities in the solution, including those from fission, material impurities, and corrosion products. The final product also goes through several quality control processes to ensure the product meets all of the requirements for purity and concentration.

The uranium target solution itself is recycled many times in the process as only a very small amount of it is consumed throughout the course of a year. The recycled solution is monitored, purified, and/or adjusted as needed prior to being returned to the subcritical assembly. Liquid wastes from the isotope production processes are handled by SHINE's waste management systems. SHINE's waste management program and systems are used to properly classify, treat, and process waste streams from all of the isotope production processes. The final isotopic composition, including all fission products, corrosion products, and other constituents of the waste stream, is used to determine the classification in accordance with the NRC low-level waste regulations.

Contamination will be controlled throughout the SHINE production facility using standard industry best practices, including contamination control work practices. The uranium solution will be confined within process systems, and within piping and tanks. Hotcells and shielded pipe trenches provide secondary confinement of the solution. Cascading HVAC zones prevent airborne contamination. Various confinement and source reduction techniques will be used throughout the plant, including gloveboxes, protective clothing, pipe flushing, and good housekeeping.

QUESTION 12 [addressed to Greg Piefer, SHINE]: In theory, at least another method to produce Mo-99 is to irradiate Mo-100 with protons. Had you looked into that method? If so, how does its cost/benefits compare with the method SHINE has chosen?

ANSWER 12 [answered by Greg Piefer, SHINE]: SHINE continuously looks at other methods to produce Mo-99, both to ensure SHINE is choosing the most technologically promising path and to assess potential competition. However, SHINE did not choose to produce Mo-99 by irradiation of Mo-100 with protons. This method would be extremely costly. Based on the National Resource Council's *Medical Isotope Production Without Highly Enriched Uranium*, previous exploration into the production of Mo-99 from the ^{100}Mo(p,pn)^{99}Mo reaction indicated a thick target yield (40-45 MeV) of 3.8 mCi/μAh. The daily production for a cyclotron would be about 50 Ci thus about 100 cyclotrons would be required for this approach to meet the U.S. demand.

QUESTION 13 [addressed to Greg Piefer, SHINE]: What is the quantity of Mo-99 that SHINE will produce?

ANSWER 13 [answered by Greg Piefer, SHINE]: The SHINE facility will be licensed to produce up to 8200 6-day curies of Mo-99 per week. SHINE will produce what is commercially demanded, up to that level.

QUESTION 14 [addressed to Greg Piefer, SHINE]: Have you done a full-scale demonstration, or bench scale? If so, how much Mo-99 has been produced?

ANSWER 14 [answered by Greg Piefer, SHINE]: Full scale demonstrations have been done on each component of the SHINE process, or on a substantially similar platform (accelerator output, irradiation of solution, separation and purification of target, off-gas handling and waste-handling). However, a fully integrated system has not been tested, and cannot be tested until an NRC license is granted. SHINE is confident that integration of the neutron driver with the uranium target will produce the expected yield, and the remaining purification processes are demonstrated.

QUESTION 15 [addressed to Parrish Staples, DOE]: Do any of the proposed medical isotope facilities pose any proliferation risk due to the new technology (SHINE), materials used, or materials remaining after extraction of the Moly-99?

ANSWER 15 [answered by Parrish Staples, DOE]: While each technology being pursued by commercial entities in the United States has unique characteristics, all will operate without the use of HEU. In fact, some of them will not use any uranium in their production processes. This is consistent with NNSA's efforts to minimize the use of HEU in civilian applications worldwide.

QUESTION 16 [addressed to Parrish Staples, DOE]: Have you done a full-scale demonstration, or bench scale? If so, how much Mo 99 has been produced?

ANSWER 16 [answered by Parrish Staples, DOE]: The conversion of the domestic high performance research reactors (HFIR, ATR, NBSR, MURR, MITR) from highly enriched uranium (HEU) fuel to low enriched uranium (LEU) fuel is not anticipated to impact the production of Mo-99. The NNSA's Reactor Conversion Program works with each facility to ensure that the necessary performance characteristics of the reactors are maintained in the conversion from HEU fuel to LEU fuel.

Although these facilities could conceivably be irradiators of LEU targets or utilize alternative technologies to produce Mo-99, they would be required to meet the exemption as specified in the American Medical Isotopes Production Act of 2012. In addition, commercial industry would have to initiate any effort to utilize such a facility for production of Mo-99.

There are a number of entities pursuing Mo-99 production in the United States using a variety of technologies that range from dedicated isotope production reactors to linear accelerators. Some entities are pursuing commercial domestic Mo-99 production without assistance from the NNSA. Given the level of commercial interest in production, the U.S. Government cannot produce Mo-99 as it would be competing against commercial industry and be in inconsistent with stated U.S. national policy.

T10 Operating Experience with Nondestructive Examinations of Nuclear Power Plant Components

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Questions submitted during the above session were answered during the session's Q/A period.

T11 Safety Culture Journeys: Lessons Learned from Culture Change Efforts

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Session Coordinator: Stephanie Morrow, Safety Culture Program Manager, Office of Enforcement, 301-415-1986, Stephanie.Morrow@nrc.gov

QUESTION 1 [addressed to Phillip Summers, TVA, Richard Haug, OPPD, and Michael Annacone, CB&I]: How do you ensure safety culture appreciation and understanding gets aligned from senior management all the way through first line and individual contributors? How to measure? Role of first line supervisors?

ANSWER 1 [answered by Phillip Summers, TVA]: Browns Ferry Nuclear (BFN) utilizes a spectrum of tools to verify alignment within the organization on understanding the importance and significance of a strong nuclear safety culture. These would include, for example, paired observations, 360 degree assessments, pulse surveys, one-on-one meetings, leadership assessments, and normal interactions with employees.

ANSWER 1 [answered by Richard Haug, OPPD]: Fort Calhoun Station (FCS) addressed alignment through the use of formal training to all leadership including Senior Leadership Team (SLT), Middle Managers and First Line Supervisors. We also gave training to the entire site individual contributors. We also give annual refresher training to the entire leadership team. Every week we emphasize safety culture and monthly our safety culture pulse survey results at the Weekly Leadership Alignment meeting. This information is then rolled out to the entire site the following day. We have SLT members attend these roll out meetings to ensure the message is consistent and given with the correct passion. These rollout meeting are given by a mixture of first line leaders and middle managers.

ANSWER 1 [answered by Michael Annacone, CB&I]: Given the unique nature of our business, we started with very clear policy statements from the CEO regarding nuclear safety culture (NSC) and safety conscious work environment (SCWE). We assigned functional responsibility across multiple business lines for NSC through a single nuclear safety officer to ensure effectiveness and consistent implementation. We are implementing initial and routine training with consistent content across all business lines with practical exercises and group sessions to allow leaders to apply knowledge. Leadership alignment meetings and leadership assessments have been used in targeted locations based upon performance for gap closure. NSC Monitoring panels and communications of results, Pulse surveys, formal NSC assessments and surveys are used to monitor and assess alignment through the results we are getting. Similarly CAP trend codes for NSC are used. An executive nuclear safety council provides oversight of the implementation of NSC related activities as well as employee concern program (ECP) information to identify trends, gaps, and common issues across business lines and ensure leaders are engaged and taking appropriate actions. Measures revolve around pulse survey results, corrective action program (CAP) trend codes, ECP program data. First line supervisors' role is to establish an environment where workers are accountable to perform work safe and error free and to feel free to stop work and raise concerns.

QUESTION 2 [addressed to Phillip Summers, TVA, Richard Haug, OPPD, and Michael Annacone, CB&I]: Given the importance of leadership in correcting a poor safety culture and sustaining a strong safety culture, what are the three most effective things your organization did to get your leaders on board and be part of the solution vice the problem?

ANSWER 2 [answered by Phillip Summers, TVA]: The three most effective actions taken by BFN to ensure that members of the senior leadership team fully internalized the importance of a strong nuclear safety culture and were fully engaged in improving that culture included (1) implemented Quarterly FAST Feedback sessions for all first line supervisors and above (forcing function to require delivery of coaching and feedback in areas that were directly linked to the Traits for a Healthy NSC), (2) implemented paired observations with the Site Vice President, Plant Manager, and all of their direct reports, and (3) altered the conduct and construct of the nuclear safety culture monitoring panel (NSCMP) meeting to include significantly more time (e.g., 2.5 hrs vice 1 hr) for discussion and challenge, and broadened and expanded the required attendee list at the NSCMP to include additional senior leaders that otherwise would not normally attend this meeting.

ANSWER 2 [answered by Richard Haug, OPPD]: First thing we did was to get the SLT aligned through some off-site team building sessions. Second item was to complete a leadership assessment of all leadership positions to identify any gaps in their ability to be a leader and created individual improvement plans for all leaders. The third item was to measure our safety culture monthly and give department specific results and hold the leaders of those departments accountable for their department results and have them create safety culture action plans if needed.

ANSWER 2 [answered by Michael Annacone, CB&I]: 1. Define expectations for leader behaviors/support of NSC/SCWE. 2. Leader assessments and Routine alignment meetings. 3. Oversight of performance metrics to ensure leader engagement and accountability in NSC/SCWE/ECP.

QUESTION 3 [addressed to Phillip Summers, TVA, and Richard Haug, OPPD]: How did you restore the trust with the public (your neighbors)?

ANSWER 3 [answered by Phillip Summers TVA]: Despite the recent regulatory challenges at BFN, there was not a discernible decline in the trust of the neighboring public confidence, and as such, no additional actions were required to be taken. That said, the BFN leadership team, as well as corporate Tennessee Valley Authority (TVA) executives, participated in quarterly public meetings in order to understand any new or continuing concerns on the part of the public.

ANSWER 3 [answered by Richard Haug, OPPD]: Fort Calhoun did not have a large reduction in the public's trust, but that was not taken for granted, so we schedule public meetings with the neighboring communities in a town hall format. We also had several onsite opportunities for the press and the public to visit the station and see with their own eyes what was occurring and to be able to ask questions. Our overall goal was transparency of our recovery with our customer owners since we are a public utility.

QUESTION 4 [addressed to Richard Haug, OPPD]: What is the more significant regulatory affairs lesson learned regarding a utility recovering from NRC degraded nuclear safety cornerstones?

ANSWER 4 [answered by Richard Haug, OPPD]: Some of the lessons learned were the need to take a very hard look at your organization through an independent assessment of safety culture. Then determine a way you can measure and check/adjust any needed corrective actions. Any regulatory issue needs to be fully analyzed, responded to with corrective actions and comprehensive closure of those actions, and lastly have specific, measurable, achievable, realistic and timely effectiveness measures. In simple terms the corrective action program is your friend.

QUESTION 5 [addressed to Richard Haug, OPPD]: If "security" is low, what is the relative impact on cyber security?

ANSWER 5 [answered by Richard Haug, OPPD]: No discernable impact on cyber security. The security organizations safety culture issues are more about a respectful work environment and ensure they recognize they are part of the overall station team.

QUESTION 6 [addressed to Diane Sieracki, NRC]: How does or will the safety culture program help inform inspectors in identifying cross-cutting aspects of violations and findings in the ROP?

ANSWER 6 [answered by Diane Sieracki, NRC]: The recently revised inspection manual chapter (IMC) 0310 incorporates the results of the safety culture common language initiative into the descriptions of the cross-cutting aspects. Inspectors should determine if the most significant causal factor of a performance deficiency matches any of the revised descriptions of the cross-cutting aspects defined in IMC 0310. NRC staff also recently issued the document NUREG-2165, "Safety Culture Common Language." It captures the safety culture common language for use in all NRC programs, including examples. In deciding which aspect is most appropriate to assign to a performance deficiency, inspectors can refer to the attribute examples provided in NUREG-2165. Additionally, NRC inspectors and management were trained on the revised guidance and its implementation in December 2013, and the Office of Nuclear Reactor Regulation is working with the NRC's Technical Training Center to develop new training courses on safety culture for inspectors.

QUESTION 7 [addressed to Diane Sieracki, NRC]: The ROP safety culture approach was not effective in identifying the safety culture shortfalls at FCS. What is being done to evaluate the cross-cutting concerns approach to make it more relevant in identifying safety culture concerns?

ANSWER 7 [answered by Diane Sieracki, NRC]: Staff is conducting an effectiveness review of the substantive cross-cutting issues (SCCI) process as part of the reactor oversight process (ROP) enhancement project. Based on the results of that review, and input from stakeholders, staff is considering revisions to the process. Industry has also provided a model of an alternative to the SCCI process that uses the safety culture assessment process described in the NEI document 09-07, "Fostering a Strong Nuclear Safety Culture," in concert with the licensee corrective action programs in assessing licensee safety culture. A working group has been established to evaluate the industry proposal to determine if it is a viable option to replace the SCCI process.

T12 Understanding the Need and Effectiveness of Remediation involving Non-routine Radionuclide Releases from Nuclear Facilities

Session Chair: Thomas Nicholson, Senior Technical Advisor, Division of Risk Analysis, RES/NRC, 301-251-7498, Thomas.Nicholson@nrc.gov

Session Coordinator: David Aird, Reliability and Risk Engineer, Division of Risk Analysis, RES/NRC, 301-251-7926, David.Aird@nrc.gov

Questions submitted during the above session were answered during the session's Q/A period.

TECHNICAL SESSIONS
Wednesday, March 12, 2014, 1:30 p.m. – 3:00 p.m.

W13 Considerations for an Enhanced Safety and Security Defense-in-Depth Strategy during Hostile Action Events

Session Chair: Brian McDermott, Deputy Office Director, NSIR/NRC, 301-287-3734, Brian.McDermott@nrc.gov

Session Coordinator: Carolyn Kahler, Emergency Preparedness Specialist, Division of Preparedness and Response, NSIR/NRC, 301-287-3722, Carolyn.Kahler@nrc.gov

Questions submitted during the above session were answered during the session's Q/A period.

W14 Current Activities in International Research–Part 1

Session Chair: Steven West, Deputy Office Director, RES/NRC, 301-251-7400, Steven.West@nrc.gov

Session Coordinator: Lisa-Anne Culp, International Relations Specialist, International Programs Team, RES/NRC, 301-251-7672, Lisa.Culp@nrc.gov

Questions submitted during the above session were answered during the session's Q/A period.

W15 Digital Instrumentation and Controls: Considerations of Embedded Digital Technology in Plant Equipment

Session Chair: Ian Jung, Branch Chief, Division of Engineering, NRO/NRC, 301-415-2969, Ian.Jung@nrc.gov

Session Coordinator: Dinesh Taneja, Electronics Engineer, Division of Engineering, NRO/NRC, 301-415-0011, Dinesh.Taneja@nrc.gov

The questions below were not answered during the above session.

QUESTION 1 [addressed to the Panel]: Need to address concept of non-relevant EDDs. There are EDDs at such an embedded level that they become a "given" part of a device design. I2C and the EDD are / should be treated the same way as C Code. De facto standards to all. Ex: 4-line 80 character display using I2C buss.

ANSWER 1 [answered by Bernard Dittman, NRC]: The commenter used the phrase "non-relevant EDDs." This is understood to refer to devices with embedded programming of limited and well-specified functionality that cannot be modified by the end-user. The example, I^2C, which stands for "Inter IC" or "Inter Integrated Circuit" is equated to "standard C code."

The C language has an ANSI standard that includes the specification of standard library functions. C compilers include vendor or 3rd party supplied code to implement the standard library functions. These libraries represent the only "standard C code." Each library function has limited functionality, a specified interface, and cannot be modified by the end-user. Furthermore, suppliers of C compilers with standard C libraries subject their compiler to a certification before they can claim it is compliant to the ANSI Standard for C. Developers of nuclear power plant safety systems that embed standard C library functions provide configuration control for the compiler and library code and include this code within the overall verification and validation effort.

Unlike the C language, the I^2C specification is controlled by a single company (NXP), who retains all rights and includes legal disclaimers on the information the specification provides. The wide-use of this specification causes it to be referred to as a "de facto standard." However, it is not a standard. Rather, I^2C is a controlled specification that is periodically revised. The I^2C

specification supports the interconnection of integrated circuits (on board and between boards) using a serial interface. The interconnection is commonly referred to as an I^{2}C bus. Integrated circuits that have an I^{2}C interface must embed firmware that is compliant with the bus specification. Similar to C library functions, when a manufacturer embeds an I^{2}C interface in its device, it may either develop its own I^{2}C programming or use 3rd party intellectual property. Also like C library functions, end users of 3rd party intellectual property do not modify this firmware, and end users of an integrated circuit with an I^{2}C interface cannot modify its firmware. Similar to ANSI standard C compilers, manufacturers of commercially available integrated circuits with I^{2}C firmware subject their devices to a certification of compliance to a defined revision of the I^{2}C specification. Similar to use of standard C library functions, developers of nuclear power plant safety systems that rely upon devices with I^{2}C interfaces should provide configuration control for the devices and include them within the overall verification and validation effort.

Therefore, the comparison of standard C library code to the embedded I^{2}C interface firmware within a commercially available integrated circuit reveals four important properties: 1) limited functionality, 2) controlled specification, 3) subject to certification of compliance, and 4) cannot be modified by the end-user. When these properties are maintained, firmware like that to implement an I^{2}C bus may be treated similar to standard C library code. This treatment includes its configuration control, assessment of quality and suitability for use, and inclusion in verification and validation efforts to provide assurance that the behavior (faulty or allowed by specification compliance) will not adversely impact the performance of an overall system safety function.

QUESTION 2 [addressed to the Panel]: Currently AP-1000 non-nuclear grade valve actuation are requiring nuclear grade digital technology that is e.g. IEEE/ QME ratings, N stamp. Can you shed some light as to why and will this be the norm going forward with new construction.

ANSWER 2 [answered by Dinesh Taneja, NRC]: It is not clear as to what specific component and the associated application this question is referring to. Some of the non-safety related SSCs in plants with passive safety systems do get additional regulatory treatment and are classified as RTNSS (Regulatory Treatment of Non-Safety Systems). For new construction, 10 CFR 50.69 provides the framework for risk-informed safety classifications. Please refer to US NRC Regulatory Guide 1.201 for additional guidance in this area.

QUESTION 3 [addressed to NRC and Utility]: Since EDDs are now out in the plant in Mods affecting mechanical, electrical, etc. in safety or non-safety. How do we make sure that we know by utility I&C and NRC that the Mod includes EDDs?

ANSWER 3 [answered by Bernard Dittman, NRC]: There is no regulation that requires nuclear power plant licensees to identify all plant equipment (safety and non-safety)—whether original or implemented in modifications—that includes an EDD, and report this information to the NRC. However, existing regulations require nuclear power plant licensees to comply with the applicable regulations, licensing basis, and where applicable the Appendix B requirements, which includes configuration control of plant equipment and testing to demonstrate suitability for use in the installed environment. Additionally, all changes to the nuclear power plant must comply with 10 CFR 50.59 or Section VIII of the relevant design certification rule. These regulations allow a licensee to modify its plant without prior NRC approval in certain circumstances. The NRC staff may inspect the licensee's evaluations performed to demonstrate compliance with 10 CFR 50.59 or Section VIII when the licensee makes a change without prior NRC approval. When doing so, the staff can consider the existence of EDDs. When a modification requires prior NRC approval, the NRC staff considers the existence of EDDs during the safety evaluation that supports the corresponding license amendment. NEI 01-01 provides guidance for performing digital upgrades and is being updated to address digital upgrades issues, including the use of EDDs.

QUESTION 4 [addressed to Bill Catullo, WEC]: On non-safety DCS upgrades like PWR NSSS controls, is there a standard or guidelines to use or follow?

ANSWER 4 [answered by Bill Catullo, WEC]: The various codes and standards that guide classification of Structure, System, or Component (SSC) are inconsistent in their approach and terminology when applied to I&C classification. One of the standard I suggest to use during development of one's safety classification for an SSC would be to consider the functional categorization criteria in IEC 61226. Since NSSS control systems play a vital role in establishing the initial conditions for Chapter 15 events, they should be considered important to safety systems. This recommendation is based on the fact that these control systems play an important role maintaining main process variables within the limits assumed in the safety analysis and whose failure could lead directly to operation of a protection function . For existing analog controls, single loop integrity (with the exception of power supply failure scenarios) for these control systems was assumed in the safety analysis for initial conditions (that is, only one control loop would fail at a time) or alarms would be present to alert the operators to a departure of a control function from its operating band. For integrated digital system replacements, one needs to consider the fact that potential common cause failures or system interactions between the NSSS control systems could occur. For EDDs, undetected software errors could also have an effect on system interactions that were not originally considered when establishing credible failure modes. These types of new failure modes when upgrading from analog to digital components/systems need to be addressed to ensure that the plant is not subject to an unanalyzed conditions (at a minimum).

QUESTION 5 [addressed to Mark Bowell, ONR, UK]: Can you comment as to why France is not part of the international task force? (you have Canada, UK, Finland, Germany, …., not France)

ANSWER 5 [answered by Mark Bowell, ONR, UK]: Historically, France has been a valuable member of the task force and has contributed to earlier versions of our common position document. Most recently, ASN participated in our meeting in March 2010. We understand that ongoing involvement has been prevented by resource constraints but would very much welcome France joining with us once again. If there is anything the task force can do to help facilitate this please let me know.

QUESTION 6 [addressed to Mark Bowell, ONR, UK]: What types of environments have you approved for applications? Temperature; Radiation; other?

ANSWER 6 [answered by Mark Bowell, ONR, UK]: I am not aware of any nuclear assessments in the UK for the use of embedded digital devices in harsh environments. In general, an assessment will either be aligned with the manufacturer's environmental specification in the datasheet or will involve type testing against an environmental specification defined by the licensee (which is usually also within the manufacturer's specified limits).

W16 License Renewal beyond 60 Years

Session Chair: Melanie Galloway, Deputy Director, Division of License Renewal, NRR/NRC, 301-415-1183, Melanie.Galloway@nrc.gov

Session Coordinator: Evelyn Gettys, Project Manager, Division of License Renewal, NRR/NRC, 301-415-4029, Evelyn.Gettys@nrc.gov

The questions below were not answered during the above session.

QUESTION 1: One staff recommendation for SLR entails requiring time in the PEO to acquire operating experience to inform SLR before allowing an application. It seems the 40 yrs the current regs require already provide much more OE to inform SLR than the 20 years of operating in the initial license renewal, e.g., 2/3 of the requested operating term vs. ½ for the initial license renewal. Comments?

ANSWER 1 [answered by NRC]: The staff's recommendation for additional time in the PEO is to acquire operating experience and lessons learned specifically from the implementation of the Aging Management Programs (AMPs) in the first license renewal. The intent of this recommendation is to ensure that any applicant for SLR is able to adequately assess the effectiveness of how the AMPs were implemented for the first renewal. While the first 40 years of operation does afford much operating experience that can be incorporated into future programs, the requirements of the first 40 years of operation do not specifically require AMPs in the context of license renewal (i.e., to specifically demonstrate that the intended functions of important SSCs can be maintained with these AMPs).

QUESTION 2: In Part 54, the Commission at lease considered whether renewals should be judged by a comprehensive review using all or some current standards, not just aging. Will this be considered for subsequent renewal?

ANSWER 2 [answered by NRC]: Contrary to the statement by the commenter, in its Statement of Consideration for Part 54, the Commission established the first principle of license renewal in stating that "with the possible exception of the detrimental effects of aging on the functionality of certain plant systems, structures, and components, the regulatory process is adequate to ensure that the licensing bases of all currently operating plants provides and maintains an acceptable level of safety so that operation will not be inimical to public health and safety or common defense and security" (56 FR 64946; December 13, 1991). As such, it did not indicate that license renewal required a comprehensive review using all or some current standards beyond assessment of aging effects.

QUESTION 3: We only heard the NRC is focus on safety technical issues but this plants also have an environmental impact, for example changes into population etc. Is the NRC also looking into how life beyond 60 will change the way these issues are evaluated?

ANSWER 3 [answered by NRC]: The NRC is responsible for reviewing license renewal applications for power reactor licenses in accordance with both safety (10 CFR Part 54) and environmental (10 CFR Part 51) requirements. In 1996, the NRC developed NUREG 1437, “Generic Environmental Impact Statement for License Renewal of Nuclear Plants” (GEIS) to evaluate the environmental consequences of license renewal. NRC staff issued a revised GEIS in June 2013 and believes that updated document is adequate for any subsequent renewal in the future. The GEIS covers impacts that are common to most nuclear power facilities and allows applicants and the NRC to focus on those important environmental issues specific to each site pursuing license renewal. The NRC recognized that environmental impact issues might change over time, and that additional issues may need to be considered, and as such the GEIS is periodically revisited and revised to update and reevaluate the potential environmental impacts arising from the renewal of an operating license.

QUESTION 4: Concerning the safety framework of SLR, in addition to ageing management does NRC consider safety lay-out, equipment, enhancements for plants designed in the ‘70s, in comparison with units designed in the 2000’s as AP1000?

ANSWER 4 [answered by NRC]: One of the principles of the license renewal program is to ensure that the current licensing basis (CLB) for each unit operating beyond its current operating period is maintained in the same manner and to the same extent in the period of extended operation (PEO) as it was in the previous operating period. The plants seeking license renewal have various designs of various ages. Regardless of the design or its age, the focus of license renewal is to ensure that the CLB for each unit continues to be adequately maintained in the PEO. This approach applies to licensees of currently operating reactors who have applied, and will apply, for operation beyond 40 years, operating reactor licensees that apply for operation beyond 60 years, and new reactor licensees who will apply for operation beyond their initial 40 years.

The license renewal program does not focus on the design or the age of the design. Regardless of the design or its age, the license renewal program focus is on adequate aging management of structures and components to ensure safe operation during the extended period of operation.

QUESTION 5: How is a license renewal for an old plant different from a new license delivery for a new plant? Should some Gen III safety options be used as “safety goals” for license renewal?

ANSWER 5 [answered by NRC]: License renewal for a currently operating reactor (whether for an extension for 40-60 years of operation, or for 60-80 years of operation) focuses on aging and aging management in the period of extended operation (PEO). Reviews for initial licenses are broader and consider all aspects of safe design, construction, and operation of a reactor.

Currently operating reactors have a history of plant operating experience and regulatory oversight that provides significant information on the licensee's ability to safely and effectively manage all aspects of the plant's operation. The NRC's regulatory processes have been applied to current plants over a long period and issues have been identified and addressed to ensure continued safe operation of the plant. No such practical history and demonstration has been established for new plants. Because current plants have this history and have demonstrated continued safe plant operation, the license renewal program recognizes the processes that have been used and will continue to be used (e.g., and Maintenance Program, Reactor Oversight Program, Appendix B, etc.) will be adequate for operation in the PEO. The only area where current programs have not been applied is in the identification and management of issues unique to operation in the PEO. These issues would be related to age-related degradation and the accompanying aging effects. Therefore, the NRC's license renewal program recognizes the effectiveness of current regulatory processes will continue to focus on age-related issues in the PEO.

With regard to using some Gen III safety options as "safety goals" for license renewal, such an assessment would have to be done in a comprehensive way. In particular, an assessment would have to demonstrate that application of Gen III safety options to license renewal would be needed in response to issues unique to license renewal and not needed for plants in the current operating period. As stated above, current regulatory processes are in place to make this determination. Should it be found that the Gen III safety options are needed for plants in their current operating term, these changes would become part of the CLB and carried forward into the PEO.

QUESTION 6: The first plants to seek subsequent license renewal (SLR) will be for plants with designs developed in the 1950s and 1960s. If operation beyond 60 years is approved, it will be important to use all available tools to understand operation in this new regime, including what is contributing most to plant risk. Therefore, shouldn't licenses be required to have and maintain a PRA to operate beyond 60 years?

ANSWER 6 [answered by NRC]: The staff reviewed the current regulatory framework for the license renewal program to determine what changes, if any, would be needed to the current framework to ensure continued safe plant operation in the 60-80 year period of extended operation (PEO). The current framework is based on two fundamental principles: (1) with the possible exception of age-related degradation unique to license renewal, the current regulatory process is adequate to ensure that the licensing bases of all currently operating plants provide and maintain an acceptable level of safety so that operation will not be inimical to public health and safety or common defense and security, and (2) the plant-specific licensing basis must be maintained during the PEO in the same manner and to the same extent as during the prior operating period.

The staff determined that the two principles underlying the License Renewal Program will continue to be valid for the subsequent license renewal period.

In addition, the Statements of Consideration for 10 CFR Part 54 state that probabilistic methods may be useful in helping to assess the relative importance of structures and components that are subject to an aging management review by helping to draw attention to specific vulnerabilities and may assist in developing an approach for aging management adequacy (see Federal Register 60 FR 22468, May 8, 1995). In addition, the Commission Policy statement on PRA recognizes and encourages the use of risk information in all regulatory activities to the extent practicable (see Federal Register 60 FR 42628, August 16, 1995).

Consistent with the first principle of license renewal, current processes are in place to evaluate the use of risk insights in NRC activities. The NRC has begun activities to assess how risk can be more consistently applied across all of NRC's activities and the most effective uses and applications of probabilistic risk assessments (PRAs). Such a holistic approach will ensure a more comprehensive, consistent, and stable application of PRAs in nuclear-related activities. As these assessments are completed and regulatory changes are implemented, these changes will become part of plants' current licensing bases and carried forward into the PEO.

W17 Loss of Safety Functions–Undetected Open Phase(s) in Balanced Three-Phase Offsite Power System

Session Chair: Jacob Zimmerman, Branch Chief, Division of Engineering, NRR/NRC, 301-415-1220, Jacob.Zimmerman@nrc.gov

Session Coordinator: Sergiu Basturescu, Electrical Engineer, Division of Engineering, NRR/NRC, 301-415-1237, Sergi.Basturescu@nrc.gov

The questions below were not answered during the above session.

QUESTION 1: Has industry or NRC calculated risk for various plants for open phase condition (other than Byron)? If no, any plans?

ANSWER 1 [answered by Scot Greenlee, Exelon Nuclear Generation]: No, we have not calculated risk for various plants at this point. Risk for Byron was calculated because it should be relatively bounding from a nuclear industry perspective. There are no current industry plans to calculate risk for other plants. Utilities may use risk on a case by case basis to justify any changes to industry initiative.

ANSWER 1 [answered by Roy Mathew, NRC]: No. NRC has not calculated risk for various plants based on open phase condition. Since industry has taken actions to address this design vulnerability, NRC has no plans to calculate the potential risk at other plants.

<u>W18</u> **<u>The Promises and Perils of Risk-Informed Decisionmaking</u>**

Session Chair: Joseph Giitter, Director, Division of Risk Assessment, NRR/NRC, 301-415-2884, Joseph.Giitter@nrc.gov

Session Coordinator: Sara Lyons, Technical Assistant, Division of Risk Assessment, NRR/NRC, 301-415-2861, Sara.Lyons@nrc.gov

The questions below were not answered during the above session.

QUESTION 1 [addressed to Doug True, ERIN Engineering and Research, Inc.]: Could you give an example [of] how NRC can remove conservatism from PRA as it can be used in deciding the reduction of the 10-mile plume exposure EPZ?

ANSWER 1 [answered by Doug True, ERIN Engineering and Research, Inc.]: For current operating plants, the 10-mile EPZ is not based on PRA results. Activities are underway to risk-inform the EPZ requirements for small modular reactors (SMRs). Like any risk-informed application, realistic PRA results should provide the foundation for considering health and safety implications quantitatively, but a risk-informed process would also need to consider defense-in-depth and margins of safety.

QUESTION 2 [addressed to Doug True, ERIN Engineering and Research, Inc.]: Please discuss (point/counterpoint) the fire PRA conservatisms that Doug True alluded to in his presentation. [also see Correia response below]

ANSWER 2 [answered by Doug True, ERIN Engineering and Research, Inc.]: Activities are underway, including fire testing and operating experience reviews, to provide technical bases for addressing the areas of FPRA that can be enhanced. NRC and EPRI are working cooperatively on technical solutions to areas the industry believes may introduce conservatisms. Likewise, recent NRC work has identified some areas of non-conservatism that are being addressed.

QUESTION 3 [addressed to Doug True, ERIN Engineering and Research, Inc.]: Has a formal root-cause evaluation been conducted by industry or NRC on risk-informed applications that have not been successful versus risk-informed applications that have been successful? If not, why not?

ANSWER 3 [answered by Doug True, ERIN Engineering and Research, Inc.]: The NRC and industry have each formed senior management steering committees that are working on problem statements and potential solutions in areas where risk-informed applications have been challenging. While I anticipate that the lessons-learned from past applications will be an important input to this activity, I know of no plans for formal root-cause evaluations to be conducted.

QUESTION 4 [addressed to Doug True, ERIN Engineering and Research, Inc.]: What are the industry and NRC plans to implement a risk-informed regulatory framework? What are industry challenges? [also see Correia response below]

ANSWER 4 [answered by Doug True, ERIN Engineering and Research, Inc.]: NRC is in the process of considering risk-informed enhancements to the regulatory framework under NTTF Recommendation 1 and in evaluating the Risk Management Framework outlined in NUREG-2150. The resource requirements and relative benefits of such changes are a challenge for both the industry and the NRC.

QUESTION 5 [addressed to Doug True, ERIN Engineering and Research, Inc.]: Given your insights in using risk in a more integrated manner, what is your opinion of [the NRC's] proposed areas of improvements to SDPs? Is that really risk-informed or is it still risk-based and focused on the numbers?

ANSWER 5 [answered by Doug True, ERIN Engineering and Research, Inc.]: I believe the SDP process is a significant enhancement over past oversight processes, e.g., SALP. In general, I think that the industry believes this, too. In my opinion, there are areas for improving the integrated decision-making aspects of the SDP. For example, industry often spends significant resources on items of relatively low safety significance as part of responding to a potentially greater than green finding.

The SDP has elements that are relatively risk-based (e.g., screening of certain findings to Green). The non-Green findings are considered by the NRC's Significance and Enforcement Review Panel (SERP). The intention is that the SERP consider quantitative results as well as other factors in making the determination of all non-Green findings.

QUESTION 6 [addressed to Doug True, ERIN Engineering and Research, Inc.]: Regarding "A Return to Insights" emphasized in presentation, please give examples of prior risk-informed applications where insights versus "P" were emphasized. (e.g. IPEEE SMA)

ANSWER 6 [answered by Doug True, ERIN Engineering and Research, Inc.]: Some examples of insight-driven applications include IPEEE SMA (as mentioned in the question) as well as risk-informed ISI, the Maintenance Rule expert panels, and 10CFR50.69.

QUESTION 7 [addressed to Doug True, ERIN Engineering and Research, Inc.]: You seem to imply that PRA is opposed to conservatism and point to Fukushima as an example. Please explain how a lack of conservatism serves or protects the public. Also, why are licensees making mods based on risk credit (i.e., risk-based, such as incipient fire detection systems)?

ANSWER 7 [answered by Doug True, ERIN Engineering and Research, Inc.]: In nuclear safety, our decisions must always be conservative. However, undue conservatism in PRAs can obscure real safety drivers, divert safety focus, and, in some instances, mask risk increases. All of these are factors that do not serve the public.

Making modifications to reduce risk (e.g., to obtain risk credit) has a clear benefit to the public in that the changes are focused on making the plant safer. This is one of the key benefits of risk-informed approaches.

QUESTION 8 [addressed to Doug True, ERIN Engineering and Research, Inc.]: You state that we need to embrace and understand uncertainties and that PRA is more than mean number estimates. How do we improve uncertainty methods in PRA, particularly in Level 2 assessments, in your opinion? Is a more structured and integrated uncertainty assessment similar to LOCA methods a useful model to follow?

ANSWER 8 [answered by Doug True, ERIN Engineering and Research, Inc.]: Some uncertainties will not easily be reduced or eliminated, nor should they be ignored. The decision-making process must clearly identify key uncertainties and account for the significance of these uncertainties to the decision. Certain areas are amenable to the approaches used in other areas like LOCAs. Other areas will be much more difficult to utilize such a method. NRC and industry are continuing to work on enhancing our ability to understand the impacts of key uncertainties on decision-making.

QUESTION 9 [addressed to Doug True, ERIN Engineering and Research, Inc.]: You did not mention INPO as one of the institutions to help bring about risk-informed decision making. What should INPO be doing to help risk-informed, performance-based?

ANSWER 9 [addressed to Doug True, ERIN Engineering and Research, Inc.]: Until recently, INPO had not been engaged on risk management issues. However, over the past several years they have incorporated risk management into their oversight processes. These are constructive steps forward for both INPO and the industry.

QUESTION 10 [addressed to Doug True, ERIN Engineering and Research, Inc.]: Has the industry reviewed NRR's risk-informed decision making procedure LIC-504?

ANSWER 10 [answered by Doug True, ERIN Engineering and Research, Inc.]: LIC-504 was made public in the past few years. To my knowledge, there has been no formal review of LIC-504 by the industry. However, in my opinion, it contains many useful features that support the general thrust of my presentation.

QUESTION 11 [addressed to Richard Correia, NRC]: The focus of the NRC RES fire research seems to be on experimental testing. How does the testing inform the PRA? That is, doesn't the testing typically bound conditions and is therefore less useful in fire PRA? Can the testing contribute to fire PRA conservatism?

ANSWER 11 [answered by Richard Correia, NRC]: Performing real world experimental testing is a part of the mix in our fire research program. One of the most accurate methods to gather realistic data for the development and refinement of Fire PRA tools and methods is to perform testing that closely reflects in-plant conditions. The videos shown at the RIC highlighted a number of experiments recently performed under the RES fire research program. Ongoing or recently completed non-experimental projects include the verification and validation of computational models used to predict fire environments, the development of a methodology to perform human reliability analysis in support of fire PRA, and the elicitation of expert judgments regarding the conditional probabilities and durations of fire-induced spurious operations. RES also continues to support the development of improved fire PRA guidance.

Regarding the experimentally-oriented projects, these are typically performed for a number of reasons. Not all testing is worst case (bounding) in nature. In some cases the experimental testing is used to develop more realistic models and reduce conservatism where possible. Two examples include the Thermally-Induced Electric Failure (THIEF) model developed as a part of the Cable Response to Live Fire (CAROLFIRE) test program (NUREG/CR-6931) and the Flame Spread Over Horizontal Cable Trays (FLASHCAT) model developed as a part of the Cable Heat Release, Ignition, and Spread in Tray Installations During Fire (CHRISTIFIRE) test program (NUREG/CR-7010). The THIEF model can be used to more accurately predict electrical cable failure due to thermal insults modeled in fire PRAs. The FLASHCAT model can be used to more accurately predict fire growth and spread along cable trays for associated fire PRA scenarios. For both these cases, experimental data was used to develop improved models that reduce the conservative nature of the theoretical models previously used in fire PRAs.

QUESTION 12 [addressed to Doug True, ERIN Engineering and Research, Inc. and Richard Correia, NRC]: Please discuss (point/counterpoint) the fire PRA conservatisms that Doug True alluded to in his presentation.

ANSWER 12 [answered by Doug True, ERIN Engineering and Research, Inc. and Richard Correia, NRC]: This topic would be very challenging to address in writing and would be best discussed in an interactive meeting format.

QUESTION 13 [addressed to Doug True, ERIN Engineering and Research, Inc. and Richard Correia, NRC]: Many of the challenges we face in applying Risk-Informed Regulation stem from lack of knowledge in industry and NRC staff. What part can training and evaluation play in gaining alignment and resolving barriers to success? How should we proceed – specific action?

ANSWER 13 [answered by Doug True, ERIN Engineering and Research, Inc. and Richard Correia, NRC]: Training and education [given the context, we assume the questioner meant "education" rather than "evaluation"] are both important to the appropriate application of risk-informed methods. Broadly speaking, training provides the "how to's" for practical situations, and education provides the foundation for the training. This foundation can, for example, sensitize an analyst to the fundamental assumptions underlying the PRA tools being used, and help analysts develop solutions to problems not addressed by training or prior experience. NRC's PRA-related training activities are described in a Nuclear Energy Agency Report "PSA Knowledge Transfer," NEA/CSNI/R(2012)15 (see Section 11 of Appendix C, pages 161-177), available from http://www.oecd-nea.org/nsd/docs/2012/csni-r2012-15.pdf. (The NRC's activities include training for decision makers as well as technical staff.) EPRI also offers several training courses. In addition, EPRI and NRC offer joint training on fire PRA.

It should be recognized that in PRA applications, classroom training and education need to be supplemented by hands-on practical training (through on-the-job analyses and/or reviews) to achieve a practitioner's level of expertise.

Good training and education can, of course, help the community gain alignment when alignment is warranted. (Note that the consensus model approach of NUREG-1855 is one way to deal with differing models; the community distribution approach of NUREG/CR-6372 – the SSHAC approach – which emphasizes representing the range of views, is another.) Specific situations will likely require specific solutions. The NRC is engaging with industry at senior management levels to identify potential opportunities, challenges, and associated initiatives in implementing the NRC's PRA Policy Statement.

QUESTION 14 [addressed to Doug True, ERIN Engineering and Research, Inc. and Richard Correia, NRC]: What opportunities do you see for getting better alignment at all levels of NRC staff to allow PRA to get more use and reduce the deterministic mentality that sometimes gets in the way.

ANSWER 14 [answered by Doug True, ERIN Engineering and Research, Inc. and Richard Correia, NRC]: As with any maturing technical field (e,g., see Cornell, 1981), continued use of PRA methods, models, tools, and data in practical situations, with demonstrated value from these applications, will increase acceptance of PRA.

The NRC currently uses PRA in a risk-informed decision-making approach that uses the insights gained from the PRA results in conjunction with the results from traditional analytical tools for assessing safety in all of its nuclear reactor regulatory activities: (1) the development of regulations and guidance, (2) licensing decisions and certification of reactor designs, (3) the investigation of licensee operations and facilities, and (4) the evaluation of operational experience. Recognizing the challenge of integrating risk insights into the NRC's decision-making process, the NRC established the PRA Policy Statement that encourages the increased use of PRA methods. The NRC's risk-informed approach attempts to balance risk insights from PRAs with safety insights from deterministic analyses to assure activities at nuclear reactors are conducted safely.

It should be recognized that the traditional deterministic approach to nuclear plant licensing has helped the NRC achieve a high level of confidence in public health and safety and continues to provide an important cornerstone to safety. This is why the NRC advocates a risk-informed approach to integrated decision-making. The use of deterministic approaches in many areas of licensing is entirely consistent with the PRA policy statement which notes that existing rules and regulations shall be complied with unless these rules and regulation are revised.

NRC activities and initiatives being implemented are identified in the NRC's Risk-Informed and Performance-Based Plan. (See http://www.nrc.gov/about-nrc/regulatory/risk-informed/history/2007-present.html for a link to the current plan as well as a history of NRC activities). The NRC is engaging with industry at senior management levels to identify potential opportunities, challenges, and associated initiatives in implementing this policy.

Cornell, C.A., “Structural safety: some historical evidence that it is a healthy adolescent,” *Proceedings of Third International Conference on Structural Safety and Reliability (ICOSSAR ‘81)*, Trondheim, Norway, June 23-25, 1981

QUESTION 15 [addressed to Doug True, ERIN Engineering and Research, Inc. and Richard Correia, NRC]: Has a formal root-cause evaluation been conducted by industry or NRC on risk-informed applications that have not been successful versus risk-informed applications that have been successful? If not, why not?

ANSWER 15 [answered by Doug True, ERIN Engineering and Research, Inc. and Richard Correia, NRC]: Although no formal root-cause evaluations have been performed, there was an NRC initiative to solicit views on the success stories in risk-informed regulation and examples of risk-informed regulation that were not successful. In 2002, the NRC Advisory Committee on Reactor Safeguards (ACRS) undertook an effort to assess the NRC’s needs for improved PRA technology to risk inform its regulations. As part of this effort, NRC ACRS commissioned Karl Fleming of Technology Insights to assess issues associated with the advancement and increased use of risk information in regulatory decisions. Mr. Fleming conducted about 20 interviews with PRA practitioners and decision makers from the NRC staff and selected industry representatives including the Nuclear Energy Institute (NEI) staff. The key results of these interviews are contained in NUREG/CR-6813, “Issues and Recommendations for Advancement of PRA Technology in Risk-Informed Decision Making,” April 2003. The insights from the ACRS and NRC staff interviews provide a snapshot of both successful and unsuccessful examples of risk-informed regulation at that particular time. In the NUREG/CR-6813 report, the author provided recommendations to advance the capability of PRAs to support risk-informed decision making.

Since promulgation of the NRC PRA Policy Statement in 1995, NRC has completed the development of many regulatory guidance documents (e.g., Regulatory Guide 1.200), and continued ongoing activities (e.g., PRA Standards development) to promote the use of PRA technology in risk-informed regulatory applications. The use of these regulatory guidance documents and other risk-informed tools by NRC and industry staff speaks for the successful use of PRA technology in risk-informed regulatory applications.

The identification of lessons-learned regarding risk-informed applications is an idea worth considering (e.g., by industry and NRC committees considering how to address issues associated with current and future applications). It should be cautioned that: (a) “success” (and “failure”) has multiple dimensions, (b) different stakeholders may value these dimensions differently, (c) the approach taken for specific licensee applications may involve complex, idiosyncratic factors, and (d) the effects of a particular application may not be fully felt for some years. Thus, the analysis may not be straightforward.

QUESTION 16 [addressed to Doug True, ERIN Engineering and Research, Inc. and Richard Correia, NRC]: What are the industry and NRC plans to implement a risk-informed regulatory framework? What are NRC challenges?

ANSWER 16 [answered by Doug True, ERIN Engineering and Research, Inc. and Richard Correia, NRC]: The NRC has not yet decided on whether or how to make changes to its risk-informed regulatory framework. In December 2013, the NRC staff provided the Commission with recommendations for improving the NRC's regulatory framework in SECY-13-0132, "U.S. Nuclear Regulatory Commission Staff Recommendation for the Disposition of Recommendation 1 of the Near-Term Task Force Report" (ADAMS Accession No. ML13277A413). These recommendations are in response to the Near-Term Task Force recommendation to establish a logical, systematic, and coherent regulatory framework for adequate protection that appropriately balances defense-in-depth and risk considerations. Also, as directed by the Chairman's Memorandum, "Evaluating Options Proposed for a More Holistic Risk-Informed, Performance-Based Regulatory Approach," dated June 14, 2012 (ADAMS Accession No. ML121660102), the staff is developing a response to the recommendations in NUREG-2150, "A Proposed Risk Management Regulatory Framework". Both these activities are being closely coordinated.

On May 19, 2014, in its staff requirements memorandum on Near-Term Task Force Recommendation 1, the Commission directed that the objectives of certain regulatory framework improvement activities recommended in SECY-13-0132 be deferred to be reevaluated, as appropriate, within the context of the Commission direction to be provided on the long-term Risk Management Regulatory Framework (RMRF) described in NUREG-2150. The RMRF SECY paper is scheduled to be provided to the Commission in November of 2014.

QUESTION 17 [addressed to Anil Julka, NextEra Energy]: Should SPAR be used as a tool for generically requiring filtered vents or as a tool to validate licensee specific vent strategies?

ANSWER 17 [answered by Anil Julka, NextEra Energy]: No – licensees should use their plant specific PRA models for risk assessments. SPAR is a simplified PRA tool that the NRC uses to independently validate the licensees risk evaluations for license amendment requests, NOEDs, etc. Also, the NRC uses SPAR as the basis for their Significance Determination Process (SDP) evaluations for identified plant performance deficiencies. The NRC developed and maintains the SPAR model. The licensees must use their plant specific PRA models for plant risk evaluations and are expected to develop and maintain these models to the standards defined by the NRC in RG 1.200. The licensees should assure that the SPAR models are up to date (i.e. reflect the as built and as operated plant), have valid assumptions, and that they understand the differences in the assumptions used in SPAR as compared to their plant specific PRAs. To this end the licensee should promptly communicate any changes to their PRA models to the NRC.

QUESTION 18 [addressed to Anil Julka, NextEra Energy]: Please speak to the issue of voluntary entry into LCOs. Should SPAR or other PRA models be used to determine whether this is a detriment to safety?

ANSWER 18 [answered by Anil Julka, NextEra Energy]: Yes – licensees should, and do, assess changes to risk for voluntary entry into LCOs. As required by 50.65(a)(4), licensees must evaluate the risk of changes to a plant configuration prior to entering that configuration. Typically, and likely universally, the licensees use a risk monitoring tool to perform this risk assessment. The risk monitoring tool uses the plant specific model or is based on rules that consider results based on that model. SPAR will not be a good application for online risk monitoring functions since several two unit sites have different alignments that are not captured in SPAR. NRC may be able to provide further insights into the SPAR.

QUESTION 19 [addressed to Sunil Weerakkody, NRC]: What is NRC doing to update SDP risk models/process? Gaps are widening between licensee and NRC risk analysis results and is having a significant impact on findings due to excessive conservatisms when compared to licensee results.

ANSWER 19 [answered by Sunil Weerakkody, NRC]: The NRC takes several steps to ensure that SPAR models and associated risk assessments that support the SDP process are of high quality and reflect the as-built, as-operated plants. These measures include:

- Implementation of a SPAR Model Quality Assurance Plan
- Use of the Risk Assessment of Operational Events Handbook (also known as the "RASP Handbook) by agency risk analysts. This Handbook helps to ensure consistency in the application of risk assessment methods and tools. Additionally, Volume 3 of the Handbook provides detailed guidance on performing SPAR model modifications and reviews to ensure that the models are of adequate quality and reflect the as-built, as-operated plant for the problem being analyzed.
- Comparisons between results using the SPAR baseline model results and licensee model results (when voluntarily submitted by the licensee) to identify differences and resolve issues as appropriate. These comparisons include comparisons of baseline CDF, conditional core damage probability for each initiator type, and top cut sets.
- Periodic updates of SPAR models. The NRC performs major updates on approximately 8-12 SPAR models per year based on feedback from Regional Senior Reactor Analysts and utility PRA analysts. Additionally, less significant model changes to roughly 20 models per year are performed to support risk assessments for specific regulatory applications. Therefore, approximately a third of the plant-specific SPAR models are typically updated in a given year.
- Senior Reactor Analysts must complete a rigorous training and qualification program (described in Inspection Manual Chapter 1245, Appendix C-9) to ensure that they can proficiently apply risk tools in Reactor Oversight Process (ROP) applications. Each SDP analysis is peer reviewed by another SRA or Headquarters analyst.
- Significance and Enforcement Review Panel (SERP) reviews are held for all "greater than green" findings to ensure appropriate risk assessment methods are applied with consistency in the ROP. The SERP includes headquarters staff from the Office of Enforcement, the applicable Region, and NRR technical, program, and management representatives.
- As part of the Enforcement Process, licensees are afforded an opportunity to either attend a Regulatory Conference or provide a written response for all performance issues of "Greater than Green" significance. The NRC will conduct a review of all information

provided by the licensee to ensure it is appropriately considered in the agency's risk assessment. IMC 609, Attachment 2, also discusses the ground rules of an appeal process for a licensee to appeal the staff's final SDP assessment.

All of the above described measures taken together ensure that the final agency risk assessment accurately reflects the performance issue and the as-built, as-operated plant configuration. However, it is important for licensees to provide relevant, accurate, and timely information in support of this process.

With regard to the (questioner's) comment that there is a widening gap between SPAR and licensee risk models, the NRC's experience has not validated this concern. Differences between results from SPAR model results and licensee PRA models tend to be driven by the characterization of the event and specific assumptions for the analysis (e.g., common cause failure potential, exposure time, success criteria, recovery credit, and physics of failure), rather than differences in the baseline risk models. For this reason, the agency ensures that all NRC risk analysts supporting the ROP/SDP activities are thoroughly trained and implement many process controls (as discussed in IMC 0609 and the RASP Handbook) to ensure that results obtained from agency risk assessments are reliable, transparent, and predictable.

QUESTION 20 [addressed to Sunil Weerakkody, NRC]: I agree that most differences between SPAR and NRC models are driven by assumptions/ boundary conditions. Even when the results are closely aligned, small differences can result in different colored finding due to fairly hard thresholds. In this case, shouldn't the model which provides the best alignment with the as-built, as-operated plant be used?

ANSWER 20 [answered by Sunil Weerakkody, NRC]: As described in the response to the previous question, the NRC implements numerous measures to ensure that risk assessments are reliable, transparent, predictable, and reflective of the as-built, as-operated plant configuration. Additionally, the agency follows a deliberative process (as described in IMC 0609 and IMC 612) to characterize licensee performance issues. This process includes the opportunity for licensees to provide additional information to support the staff's risk assessment and also provides an appeal process. Furthermore, it should be noted that PRA models are very complex representations of the nuclear plant response to postulated initiating events and conditions. Although the SPAR models are plant-specific models, they rely on a set of standardized modeling conventions (e.g., standardized naming conventions and logical structure) to allow agency risk analysts to proficiently assess the risk significance of findings and operational events. Even if licensee PRA models were to be made available to the NRC staff, it would take an undue amount of agency resources to verify and understand all licensee modeling conventions to ensure that operational events and findings were accurately modeled in these PRAs. Therefore, to ensure the enforcement process remains objective and unbiased, the NRC relies on independent risk assessments performed by NRC analysts using the SPAR models, rather than the licensee's risk characterization. With regard to situations where the NRC and licensee risk characterization results are close but straddle a threshold, it is rarely obvious which model provides the "best alignment with the as-built, as-operated plant." However, the collective quality assurance measures described in the response to the previous question ensure that the final NRC risk characterization is realistic and timely based on the best available information. Additionally, internal NRC reviews, such as the SERP, will specifically consider such factors to ensure that the final risk characterization is appropriate.

QUESTION 21 [addressed to Sunil Weerakkody, NRC]: Regarding SDP SERP process, why is the licensee input to the process so informal? NRC region SRA “informally” solicits licensee input, passes it to NRR SDP lead PRA and input only needs to be “considered” with no dialog with licensee PRA staff. Getting more formal involvement of licensee PRA staff would improve dialog and potentially avoid reg conferences.

ANSWER 21 [answered by Sunil Weerakkody, NRC]: As part of the inspection process prior to entry into the SDP, NRC inspectors and Regional SRAs will attempt to gather all relevant information that is pertinent to the finding. The NRC follows a structured inspection, enforcement, and significance determination process when characterizing the significance of findings and potential enforcement of violations. This process is intended to ensure that the agency’s decision making process is timely, transparent, predictable, and consistent. The Regulatory Conference provides a formal venue for the licensee to provide the staff with any additional information relevant to the finding and apparent violation (as applicable) and a forum for open discussion on the influential assumptions affecting significance of the finding.

QUESTION 22 [addressed to Sunil Weerakkody, NRC]: Speaker stated that SPAR models are of high quality. These models do not meet the PRA standard requirements applied to operating plant PRAs. Why is this acceptable for SPAR, but not for the licensee model?

ANSWER 22 [answered by Sunil Weerakkody, NRC]: As described in the previous responses, the NRC implements a Quality Assurance Program to ensure that the SPAR models reflect the as-built, as-operated nuclear plant. However, it should be recognized that the primary purpose of the SPAR models is to support the reactor oversight process. While the agency endeavors to ensure the SPAR models themselves meet high quality standards, it is important that a distinction be made between model quality and the implementation of the model to support a specific risk assessment. Therefore, in order to ensure that the SPAR models are correctly applied and used for a specific application, the agency implements several measures, including use of trained staff; technical peer and management reviews of risk assessments; and opportunities for licensees to provide additional information to the staff. However, it should be noted that the SPAR models are generally used to categorize and prioritize operational events and conditions, including licensee non-compliance issues with existing regulations.

On the contrary, licensee PRA models have generally been developed to support licensing basis changes and therefore must meet more stringent technical adequacy requirements (e.g., Regulatory Guide 1.200). Although the SPAR models are not required to meet the ASME PRA standards, the NRC in 2009, with the assistance of the BWR and PWR Owner’s Groups, performed industry-led peer reviews of a typical BWR and a PWR SPAR model. The peer review teams (which consisted of representatives from ERIN Engineering, Constellation, Dominion, and Florida Power and Light) noted a number of strengths for the SPAR models, including:

- The SPAR model structure is robust and well developed.
- The SPAR model fault trees are streamlined with an appropriate level of detail for its intended uses.
- The SPAR model structure and the SAPHIRE computer software are at the state of the technology.
- The SPAR model is an efficient method to develop qualitative and quantitative insights for risk-informed applications, SDP evaluations, inspections, event assessments, and model evaluations.

The peer review teams also noted a number of enhancements that could be made for the SPAR models and the staff has been working to address these recommendations.

QUESTION 23 [addressed to Sunil Weerakkody, NRC]: There is a significant gap in the risk models of the licensee and the NRC. This has led to significant differences in risk assessment of events and equipment challenges. The impact has been significant number of NRC findings that are greater than green, such as white or yellow, where licensee analysis show green. What is the NRC doing to improve process or risk models to more accurately assess risk in the SDP. This has resulted in misapplication of important resources – both NRC and industry – in addressing issues that were not accurately characterized due to excessive conservatisms.

ANSWER 23 [answered by Sunil Weerakkody, NRC]: The NRC has not observed a "significant gap" in risk models used by the licensees and the SPAR models. While divergence between initial licensee and NRC risk assessments for a specific finding is sometimes noted, the reasons for the divergence is usually readily identified and often involves differences in key assumptions and boundary conditions rather than baseline modeling differences (see note below). Additionally, and as discussed above, the NRC implements a number of important procedural and technical measures to ensure that agency risk assessment are accurate, timely, and reflect the as-built, as-operated plant. These measures provide a high degree of confidence that ROP/SDP issues, including the risk significance characterization of licensee non-compliance with regulatory requirements, are appropriately categorized.

Note: *In this context, PRA "modeling" generally refers to the structure of the underlying logical model and includes event trees, fault trees, system success criteria, identification of common cause component groups, identification of human failure events, and recovery rules. Assumptions and boundary conditions refer to model manipulations performed by the analyst to the underlying model to represent the specific event or condition of interest (e.g., setting basic events to "true", updating data, etc.).*

TECHNICAL SESSIONS
Wednesday, March 12, 2014, 3:30 p.m. – 5:00 p.m.

W19 Current Activities in International Research–Part 2

Session Chair: Carol Moyer, Team Leader (Acting), International Programs Team, RES/NRC, 301-251-7641, Carol.Moyer@nrc.gov

Session Coordinator: Lisa-Anne Culp, International Relations Specialist, International Programs Team, RES/NRC, 301-251-7672, Lisa.Culp@nrc.gov

QUESTION 1: If plant closures in Germany are intended to eliminate threats of accidents, how then can you protect the German public from reactor accidents in neighboring countries which have many reactors close to the German boarder?

ANSWER 1: First, I would like to state that here I am representing the main German TSO, I do not represent the German regulator.

The question should much better be answered by the regulatory body.

Nevertheless, here are some elements of an answer. The protection from reactor accidents occurring in neighboring European countries has not been changed by the shutdown of NPPs in Germany.

To pursue our national interests in the highest safety standards of nuclear reactors that are developed and operated abroad the German government has been influencing the latest revision of the EU council directive for establishing a community framework for nuclear safety. Moreover, the German regulator collaborates with the major European nuclear authorities in the framework of the Western European Nuclear Regulators Association (WENRA), which e. g. established and regularly updates the so called reactor safety reference levels.

Beyond that, there is extended European collaboration in emergency management including international exercises.

GRS, as the TSO to the German regulator collaborates with other European TSOs in the European TSO Network (ETSON) that aims at the harmonization of nuclear safety practices especially in the field of safety assessment.

W20 Future Vision of Spent Fuel Storage Regulations

Session Chair: Michele Sampson, Branch Chief, Division of Spent Fuel Storage and Transportation, NMSS/NRC, 301-287-9077, Michele.Sampson@nrc.gov

Session Coordinator: Jeremy Smith, Senior Criticality and Shielding Engineer, Division of Spent Fuel Storage and Transportation, NMSS/NRC, 301-287-0928, Jeremy.Smith@nrc.gov

Questions that were not addressed during the session and their answers will be posted when they become available. In the meantime, if you have any questions pertaining to this topic, please contact the Session Chair or Coordinator listed above.

<u>W21 Nonconservative Technical Specifications? What Actions Are Needed?</u>

Session Chair: Michael Markley, Branch Chief, Division of Operating Reactor Licensing, NRR/NRC, 301-415-5723, Michael.Markley@nrc.gov

Session Coordinator: Eva Brown, Senior Project Manager, Division of Operating Reactor Licensing, NRR/NRC, 301-415-2315, Eva.Brown@nrc.gov

Questions submitted during the above session were answered during the session's Q/A period.

<u>W22 Regional Session – Contemporary Nuclear Power Plant Regulatory Issues</u>

Session Chair: Michael Johnson, Deputy Executive Director for Preparedness Programs, OEDO/NRC, 301-415-1713, Michael.Johnson@nrc.gov

Session Coordinator: Geoffrey Miller, Branch Chief, Division of Reactor Safety, RIV/NRC, 817-200-1137, Geoffrey.Miller@nrc.gov

The questions below were not answered during the above session.

QUESTION 1: If a situation like Fukushima occurred in the U.S., how would the regulators respond to the buildup of large quantities of contaminated water? Would a national U.S. waste repository help relieve this issue?

ANSWER 1: The regulators would ensure that the contaminated water was managed in a manner that protects public health and safety. The NRC would review any licensee's plans for management of the water. It would be premature to say that we would or wouldn't allow any particular approach; it would depend on the exact situation. Consistent with our principles, we would require that the risk to the public be minimized.

A national waste repository would not directly help this issue. Contaminated water is not the type of waste that is stored at a national waste repository. It's possible that the byproducts of cleaning up the contaminated water would be stored at a repository, but the water itself would not be.

QUESTION 2: In 10 years of simulator training, I witnessed three instances where crews made major errors that could have resulted in core damage, yet I saw little followup or reporting of these issues to plant management or the NRC. My questions: 1) What do licensee sites do to capture these significant simulator errors and inform management and the NRC? 2) Does the NRC need to do more regulation in this area of concern?

ANSWER 2: 1) When licensed operator crews are being evaluated in the simulator, the results of the evaluations are documented as part of their site-specific requalification training program. Performance that would result in challenging reactor safety would be evaluated as unsatisfactory performance. Licensed operators that perform in an unsatisfactory manner are typically removed from watchstanding in the plant, pending completion of a remedial training plan focused on the performance weaknesses identified. After completing the remedial training plan, the licensed operator is re-evaluated in a simulator scenario that provides the ability to evaluate if the remediation addressed the deficiencies identified. If the individual passes the evaluation, they are returned to watchstanding per the licensee procedure. If they do not pass, they are further evaluated within the licensee's requalification program process to determine if additional training is needed, or if the performance warrants a termination of the individual's operator license.

The fact that licensed operators are removed from the watch bill at a plant would result in communications from the site training department to various levels of licensee management.

The NRC learns of these performance issues in two ways: licensee notification and through the inspection process. For one, if the issue warrants termination of an individual's license, the licensee is required to notify the NRC of the need to change the licensing status of the person within 30 days of its decision (per Title 10 CFR Part 50.74). Instances where individuals fail requalification program examinations specified in 10 CFR 55.59 are reviewed by NRC inspectors as part of the scope of the annual and biennial licensed operator requalification program inspections. This is addressed in NRC Inspection Procedure 71111.11, "Licensed Operator Requalification Program and Licensed Operator Performance," Section 02.07. Licensed operator performance in the requalification program is also evaluated quarterly by resident inspectors using Section 02.11 of the same procedure.

2) The licensee's Commission-approved requalification program requires that it continually evaluate the licensed operators, identify deficiencies through systematic observation and evaluation, and document the deficiencies per 10 CFR 55.59(c)(4) and (c)(5). As part of the NRC's quarterly, annual and biennial inspections of the program's functions, we determine whether we have reasonable assurance that the program is functioning as expected. The NRC believes that these inspection programs provide adequate assurance that operating crews perform their licensed duties in a reliable manner.

W23 The Administrative Hearing Process: What Can the NRC Learn from Other Agencies?

Session Co-Chairs:

Ronald Spritzer, Administrative Judge, ASLBP/NRC, 301-415-6803, Ronald.Spritzer@nrc.gov

William Froehlich, Administrative Judge, ALSBP/NRC, 301-415-5922, William.Froehlich@nrc.gov

Session Coordinator:

Twana Ellis, Program Analyst, ASLBP/NRC, 301-415-7703, Twana.Ellis@nrc.gov

Questions submitted during the above session were answered during the session's Q/A period.

W24 What Can We Learn from Risk-Informed Licensing Initiatives?

Session Chair: Hossein Hamzehee, Branch Chief, Division of Risk Assessment, NRR/NRC, 301-415-0562, Hossein.Hamzehee@nrc.gov

Session Coordinator: Sara Lyons, Technical Assistant, Division of Risk Assessment, NRR/NRC, 301-415-2861, Sara.Lyons@nrc.gov

The questions below were not answered during the above session.

QUESTION 1: Concerning GSI-191, what is the risk significance of the issue (i.e. how much is the risk contribution to CDF?)

ANSWER 1 [answered by Hussein Hamzehee, NRC]: The risk significance of issues that are being addressed through GSI-191 is expected to vary from plant to plant due to differences in emergency core cooling systems (ECCS) and containment spray sump configurations as well as insulation types and amounts. In the staff requirements memoranda (SRM) to SECY-12-0093, the Commission approved an optional risk-informed approach for licensees responding to generic letter (GL) 2004-02 (i.e., resolving GSI-191). Licensees opting to use this risk-informed approach must quantify the portion of core damage frequency and large early release frequency attributable to debris and compare these values to the risk acceptance guidelines in Regulatory Guide 1.174. The staff is currently reviewing the application submitted by the pilot plant, South Texas Project, to determine whether the risk acceptance guidelines have been met. In addition to risk, the staff will consider other factors such as defense in depth and safety margins when determining whether GL 2004-02 has been adequately addressed for a particular plant. The initial increase in core damage frequency (CDF) reported by South Texas was 2.9 E -8 / yr (ML13323A183), although the NRC staff has not yet verified this value.

QUESTION 2: Is flex credited in PRAs and risk-informed applications? If not, are there plans to do this?

ANSWER 2 [answered by Hussein Hamzehee, NRC]: At the present time, licensees have not yet fully implemented FLEX nor has the agency stated an official position on how or if licensees can credit FLEX actions and equipment for risk-informed applications. Agency guidance on risk-informed decisionmaking (e.g., Regulatory Guide 1.174) states that PRA models should represent the as-built and as-operated plant and should quantify risk as realistically as practicable. Therefore, FLEX equipment would not be excluded from consideration in risk-informed applications; however, the degree of "credit" given would depend on several factors including, but not limited to:

- The risk-informed application in question (e.g., license amendment request, significance determination process, notice of enforcement discretion, etc).
- The degree to which the FLEX equipment or actions have been demonstrated to be viable.

It is likely the appropriate modeling of FLEX equipment and actions would need to be assessed on a case-by-case basis until sufficient operating experience has been gained.

TECHNICAL SESSIONS
Thursday, March 13, 2014, 8:30 a.m. – 10:00 a.m.

TH25 Consensus Standards for New and Improved Plants

Session Chair: Michael Case, Director, Division of Engineering and NRC Standards Executive, RES/NRC, 301-251-7619, Michael.Case@nrc.gov

Session Coordinator: Thomas H. Boyce, Branch Chief, Division of Engineering, RES/NRC, 301-251-7599, Tom.Boyce@nrc.gov

The questions below were not answered during the above session.

QUESTION 1: Could you elaborate on the extent and type of actual tests performed on steel-concrete (SC) composite material (to complement numerical simulation) in the concept validation and approval?

ANSWER 1 [answered by Amit H. Varma, Purdue University]: SC composite construction has been researched extensively in Japan, S. Korea, and the US. Large-scale tests have been performed to confirm: (i) axial compression behavior, (ii) out-of-plane shear behavior, (iii) in-plane shear behavior, (iv) combined thermal and mechanical load behavior, and (v) anchorage or connection behavior.

Please refer to the website: https://engineering.purdue.edu/~ahvarma/Research-Articles.html for a listing of US research and testing. Publications from Japan and S. Korea are also available in the public domain in Journals (Nuclear Engineering and Design by Elsevier, http://www.journals.elsevier.com/nuclear-engineering-and-design/), and SMiRT Conference Proceedings (www.iasmirt.org).

QUESTION 2: What is being done regarding conformity assessment for the performance standards that you spoke of?

ANSWER 2 [answered by Amit H. Varma, Purdue University]: Conformity assessment is not part of the standard being developed. It is up to the industry and its regulatory agency. For example, concrete structures for safety-related nuclear facilities are designed according to American Concrete Institute (ACI) 349 code. This code specifies the minimum requirements that have to be met, and conformity assessment is up to the industry and its regulatory agency.

QUESTION 3: How do standards committees communicate with each other when there is an overlap of “jurisdiction”, e.g., reinforced concrete - steel composite (RC-SC) connection design has American Institute of Steel Construction (AISC) and ACI code considerations?

ANSWER 3 [answered by Amit H. Varma, Purdue University]: There is significant overlap between the memberships of standards committees. As a result, the codes / standards committees are cognizant of their jurisdictions, and try to minimize overlap. The engineer should also design their structures to facilitate clear demarcation of jurisdictional boundaries. In addition, the current codes / standards are compatible, and the AISC standard refers to the ACI standard directly when appropriate.

QUESTION 4: We have recently seen critical, large scale defects in concrete containment structures such as Crystal River. How does the SC composite wall code inspect these structures to ensure they are defect free when manufactured, particularly at the steel-concrete interface?

ANSWER 4 [answered by Amit H. Varma, Purdue University]: Inspection of the SC composite walls can be performed using non-destructive techniques such as impact echo. Construction mockups have also been used to confirm construction and vibration techniques. Composite walls are much more tolerant of minor construction imperfections because the steel faceplates are the primary stress bearing elements. The concrete infill provides mass, radiation shielding, stiffness, and compressive strength.

QUESTION 5: High Density Polyethylene (HDPE) piping is a great path for many apps (applications). To make the applications available to more plant sites, creep analysis could be added for new plants.

ANSWER 5 [answered by Frank Schaaf, Sterling Refrigeration Corporation]: The HDPE creep mechanism is currently being researched for nuclear applications for both old and new plants.

QUESTION 6: Many of the “success stories” are more than 10 years old (Section XI Code Cases, dissimilar metal (DM) welds). What are the “emerging” success stories? Ones in the works or need more work.

ANSWER 6 [answered by Frank Schaaf, Sterling Refrigeration Corporation]: I agree with your evaluation that there are many success stories. ASME Section XI revisions and new Code Cases have been addressing and resolving industry issues since its inception. Several examples of current issues are the following:

RISK

The concept of “risk-informed” has helped solve many issues since it’s first use. A new direction for Section XI using “risk-informed” methodology is the Reliability Integrity Management (RIM) Program. To support the Generation IV plants and Small Modulator Reactors (SMRs), a new methodology is being developed. Section XI, Division 2 is being rewritten in scope to include all reactor types and sizes. This new Division will be totally "risk-informed," in that there will be no classifications by quality or risk, and only passive Structures, Systems, and Components (SSCs) will be in the program. A new Probabilistic Risk Assessment (PRA) standard has been written and published to support the RIM program.

LEAKS IN PIPING AND WELDS

Examination

Since the acceptance of Appendix VIII by the NRC in 1998, it has been revised many times to clarify and add requirements. The revisions have made it better. The Probability of Detection (POD) using Appendix VIII is now over 80%. With the success using the ultrasonic method (UT), both Section III and Section VIII now have Code Cases using UT examinations to replace the radiography method (RT) examinations that have been required for construction acceptance.

Pipe and Welds

HDPE is just one remedy for leaking buried pipe. Section XI has provided many paths to resolve issues with leaking pipes and welds. It provides methods to improve detection, evaluate the flaws found and provide acceptable repair options that restore the pressure boundary. Similar methods used for leaking buried pipe resolution were developed for Primary Water Stress Cracking Corrosion (PWSCC), including detailed nondestructive examination (NDE) examination methods for the weld overlays used for repair. This work was all done using the Code Case format.

Piping Code Cases

Below are notable code cases that can be considered "success stories"

- N-770-2 changes to incorporate optimized weld overlay and difficult to inspect large diameter cold leg temperature locations
- N-754 addresses optimized dissimilar metal weld overlys used for mitigation and repair
- N-749 Flaw Evaluation Acceptance Criteria for Ferritic Components based on Elastic Plastic Fracture Methods
- N-766 Inlay and Onlay repairs of PWSCC
- N-513-2 improved guidance in flaw evaluation procedures

QUESTION 7: Great presentation! Good success stories. Which ASME Code did not work? Why? Thank you.

ANSWER 7 [answered by Kenneth Balkey, American Society of Mechanical Engineers]: About 10 years ago, the ASME Board on Nuclear Codes and Standards looked at possibly sun-setting the ASME Committee on Qualification of Mechanical Equipment (QME). A predecessor standards committee had produced ASME N278.1-1975 (R1992), “Self-Operated and Power-Operated Safety-Related Valves Functional Specification Standard,” and ASME B16.41:1983 (R1989) “Functional Qualification Requirements for Power Operated Active Valve Assemblies for Nuclear Power Plants,” that were available to support new reactor construction. The ASME QME Committee produced ASME QME-1-1994, “Qualification of Active Mechanical Equipment Used in Nuclear Power Plants,” which included seismic and functional qualification of active mechanical equipment, including pumps and valves, to replace the earlier standards. By that time, however, all the new plants in the U.S. had been built, and the committee was down to meeting once a year by teleconference.

Fortunately, the U.S. Nuclear Regulatory Commission (NRC) representative on the ASME QME Committee informed ASME at that time that the NRC staff had been requested to update regulatory documents to support forthcoming new plant construction activities. He suggested that Regulatory Guide 1.100 titled, “Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants,” could be revised to reference an updated version of the ASME QME-1 standard that addressed the latest new plant construction needs. Given this development, the ASME QME Committee updated the standard to issue a QME-1-2007 version that was endorsed by the NRC in Regulatory Guide 1.100 with conditions related to experience-based qualification. The latest version of this standard was issued in 2012 to include new sections on standardization of experienced-based seismic equipment qualification and the qualification of dynamic restraints. It also requires that users of the standard must provide a Qualification Specification. The NRC plans to endorse this latest standard with conditions in the future.

The above situation is the closest that our ASME Nuclear Codes and Standards organization has come to sun-setting a standards effort. Given today’s current and new reactor developments, all ASME Nuclear Codes and Standards continue to be developed to meet global nuclear industry needs.

ASME also makes use of Code Cases that allow for early implementation of new technology and help to assure that requirements related to such developments meet the needs of users, thereby reducing the chance that a Code will not be utilized.

QUESTION 8: During the conference so far, there has been a lot of emphasis on the fact that every nuclear power plant is different and has different safety issues. Do you suggest that the safety requirements shall be different for each plant?

ANSWER 8 [answered by Kenneth Balkey, American Society of Mechanical Engineers]: The NRC establishes safety requirements in the Code of Federal Regulations (CFR) that generally apply to all light water reactor designs. In some cases, the requirements are unique to a particular design (but not for different plants), for example the Advanced Boiling Water Reactor and System 80+ designs are certified in Appendixes A and B to 10 CFR Part 52. The discussion that every nuclear power plant is different recognizes the inevitable variations even among the same designs because of factors such as siting, applicable requirements when the plant was licensed, construction materials and techniques, and additional systems such as emergency diesel generators. The NRC reviews the applications from individual plants to ensure that the requirements in the CFR are met, and establishes appropriate license conditions to account for such variability.

A unique feature of ASME standards is that they are not written around a particular nuclear power plant design. Rather they are written to focus on the type of reactor such as light water reactors or high temperature reactors to allow for users to develop products and services that meet the defined standards requirements. With this approach, ASME does not endorse specific products or services, and it also assures that the safety requirements are the same for each nuclear power plant design.

QUESTION 9: The lessons learned from the use of risk-informed in-service inspection (ISI) were captured in ASME Code Case N-716. Can Code Case N-716 be used for small-modular reactors?

ANSWER 9 [answered by Kenneth Balkey, American Society of Mechanical Engineers]: ASME Code Case N-716 is based on experience of risk-informed ISI programs for today's operating reactors. Current small modular reactor (SMR) designs have features that are quite different from today's plants and likely will not be able to make full use of insights from ASME Code Case N-716.

Leaders of ASME Nuclear Codes and Standards are continually meeting with SMR vendors and other cognizant stakeholders to assure that standards are in place to support this new reactor development. For example, we are currently writing an ASME Boiler & Pressure Vessel (BPV) Code Section XI Division 2 titled "Requirements for Reliability and Integrity Management (RIM) Program for Nuclear Power Plants." The idea of the new division is that current inspection requirements are not entirely applicable to newer plant designs such as SMRs and gas-cooled reactors. These new designs will need new requirements.

ASME is also hosting a Small Modular Reactors Symposium on April 15-17, 2014 in Washington DC to obtain further input from stakeholders on this important topic.

QUESTION 10: Happy to see the poster session on air operated valves (AOV) Appendix IV. Can you provide a little more detail?

ANSWER 10 [answered by Kenneth Balkey, American Society of Mechanical Engineers]: A summary of the emerging developments on the basis for the establishment of ASME Operation and Maintenance of Nuclear Power Plants (OM) Division 1, Mandatory Appendix IV Pneumatically and Hydraulically Operated Valves is provided here.

ASME OM Code Mandatory Appendix IV will be in principle similar to OM Division 1, Mandatory Appendix III Preservice and Inservice Testing of Active Electric Motor Operated Valve Assemblies in Light-Water Reactor Power Plants, with applicability to Pneumatically and Hydraulically Operated valves (AOVs and HOVs).

Appendix III is the culmination of several regulatory and industry initiatives to establish acceptable rules for design basis, preservice, and inservice testing for Motor Operated Valves (MOVs). In the early and mid-80s there were several incidents of MOVs failing to perform their design basis function, despite plants having inservice testing programs. The NRC issued Generic Letter (GL) 89-10 and requested licensees to develop and implement a program to ensure that MOVs switch settings were set such that the MOVs would perform their design basis function under design basis conditions (such as elevated ambient temperature, degraded voltage, and high differential pressure). The NRC then issued GL 96-05, due to continuing issues and program weaknesses discovered during the implementation of GL 89-10. GL 96-05 forced the plants to establish periodic verification of all safety-related MOVs to ensure proper setting for design conditions accounting for degradation of the valve. To minimize the expense of the differential pressure testing required to establish a basis for valve degradation, the majority of the industry participated in the Joint Owner's Group (JOG) MOV program. ASME OM Division 1, Appendix III establishes rules that incorporate the requirements and outcomes of GL 89-10, GL 96-05, the best practices of the industry including the JOG, and the existing inservice testing program requirements under ASME BPV Code Section XI. The NRC has endorsed ASME OM Division 1, Appendix III since 1999. Mandatory Appendix III requires plants to perform a design basis verification test, followed by a pre-service test, and then establish a periodic testing frequency for every MOV in the plant with a safety related function.

AOVs and HOVs have also had incidents similar to the issues that occurred with MOVs. A summary of incidents can be found in GSI-158. In 1999, the industry formed the JOG AOV program, similar to the JOG MOV Program. Due to the industry taking a pro-active approach, the NRC has not issued a Generic Letter similar to those issued for MOVs. Instead, the NRC issued RIS 2000-03, in which the NRC established that the preferred method of resolution is to work with the applicable industry groups to "provide timely, effective, and efficient resolution of concerns regarding POV[1] performance." The establishment of OM Division I, Appendix IV is intended to establish design testing, pre-service, and periodic testing similar to the requirements established for MOVs in Appendix III.

QUESTION 11: Please discuss RTNSS and FLEX equipment as they relate to Codes and Standards.

ANSWER 11 [answered by Kenneth Balkey, American Society of Mechanical Engineers]: RTNSS is an abbreviation for Regulatory Treatment of Non-Safety Systems and "FLEX" addresses a diverse and flexible coping capability, building on earlier safety steps, in providing an effective and efficient way to make U.S. nuclear energy facilities even safer. Per the Nuclear Energy Institute website:

"FLEX is a major step in addressing the critical problems encountered at Fukushima Daiichi: loss of power and reactor cooling capability. It provides an additional layer of backup power after an extreme event by stationing vital emergency equipment—generators, battery packs, pumps, air compressors and battery chargers—in multiple locations. Implementing FLEX will

[1] POV stands for Power Operated Valves and refers to MOVs, AOVs, HOVs, and solenoid-operated valves (SOVs)

help maintain cooling if normal systems and other backup systems fail by stationing additional pumps and power sources in multiple locations to provide cooling water to the reactors. To provide yet another layer of protection, the FLEX approach will also maintain safety even after a catastrophic event by stationing emergency equipment in secure offsite locations. The reinforcement equipment can be used if all of the plant backup equipment is insufficient."

As discussed near the end of the RIC 2014 Consensus Standards panel session, Jim Riley of NEI kindly indicated that it is intended for FLEX equipment to be non-safety related and that commercial standards would be expected to apply. The ASME Board on Nuclear Codes and Standards has a Task Force working with industry, NRC and the Japan Society of Mechanical Engineers to define standards needs following the Fukushima event in March 2011. Thus, decisions on Codes and Standards requirements for RTNSS, FLEX and other related post-Fukushima developments continue to be made on this important topic for the global nuclear power industry.

QUESTION 12: What criteria does the NRC use to guide or limit the "Exceptions and Limitations" it takes to consensus standards? If none, should guidance be established?

ANSWER 12 [answered by Thomas H. Boyce, NRC]: There are no written criteria. We appreciate the suggestion and will consider establishing written guidance. In general, the consensus standard must support the staff's regulatory positions to be considered an acceptable method for meeting the requirements of applicable NRC rules. NRC may therefore only endorse parts of a consensus standard, or place conditions on standards for a variety of reasons, including:

- the standard covers topics broader than the staff positions
- the standard provides a relaxation of safety that the staff does not believe is justified
- the NRC has provided additional or separate guidance on the topics
- the NRC is aware of unique applications or situations that are not clearly addressed by the standard

QUESTION 13: In view of the increasing globalization of the nuclear power industry, to what extent is the NRC open to approving standards that are developed outside the U.S.?

ANSWER 13 [answered by Thomas H. Boyce, NRC]: The U.S. is very open to approving such standards. The criteria for NRC use of any standard, regardless of which organization develops it, is that the standard can be used as a means to meet the requirements of applicable NRC rules and implement the staff's regulatory positions. Of particular note, as part of its development of Regulatory Guides (RGs), the staff now routinely considers harmonization with international guidance, particularly relevant guidance published by the International Atomic Energy Agency (IAEA), having documented this harmonization in over 33 final and 11 draft Regulatory Guides. As a practical matter, it should be recognized that the NRC staff may not be involved in the development of standards outside the U.S., so may be less familiar with them, making them more difficult to use.

QUESTION 14: Discuss the use of codes and standards in resolving generic issues.

ANSWER 14 [answered by Thomas H. Boyce, NRC]: Codes or standards can be an option for addressing generic issues. In general, generic issues are complex, multi-faceted issues that involve regulatory decisions to address. Therefore, codes and standards would likely be a part of a larger resolution of the issues. The end users (e.g., utilities, vendors) of the standards, standards development organizations, and the NRC would likely need to collaborate if the standards needed to be developed in a timely manner to support the resolution of the generic issues.

QUESTION 15: What branch of the NRC reviews relief requests? What is the typical NRC review time of a licensee relief request? How does the NRC prioritize the relief requests from the various licensees?

ANSWER 15 [answered by Thomas H. Boyce, NRC]: Several branches within the Office of Nuclear Reactor Regulation (NRR)/ Division of Engineering (DE) review relief requests, depending on the technical nature of the request. These branches are the Component Performance, Non-Destructive Examination (NDE) and Testing Branch, the Vessel & Internals Integrity Branch (EVIB), the Steam Generator Tube Integrity & Chemical Engineering Branch (ESGB), and the Mechanical & Civil Engineering Branch (EMCB).

Relief requests are generally considered in the order in which they are received. The typical review time is less than one year. For priority of review, relief requests are prioritized along with all other work conducted by the agency. However, if the NRC staff is informed in a timely manner that the relief requests are needed earlier than one year for a good reason, such as for an upcoming outage, NRC staff will attempt to make those requests higher priority. For most relief requests, NRC has been able to provide a safety evaluation to its licensees within one year. Exceptions have been complicated issues that may require several requests for additional information.

TH26 Power Reactor Transition from Operating to Decommissioning

Session Chair: Larry Camper, Director, Division of Waste Management and Environmental Protection, FSME/NRC, 301-415-7319, Larry.Camper@nrc.gov

Session Coordinator: Michael Orenak, Project Manager, Division of Operating Reactor Licensing, NRR/NRC, 301-415-3229, Michael.Orenak@nrc.gov

The questions below were not answered during the above session.

QUESTION 1 [addressed to Bruce Watson, NRC]: With economics driving some decisions to shutdown reactor, is there a means to shutdown, defuel, and at a later date when economics improve, restart the plant? The idea might be to have reduced costs for staffing compliance, etc. while shutdown, i.e. temporary shutdowns of 12-48 months.

ANSWER 1 [answered by Bruce Watson, NRC]: Regulations under 10 CFR Part 50 would not prohibit temporarily shutting the plant down and transferring the fuel to the spent fuel pool for an extended period, however, regulations applicable to the shutdown unit would continue to apply (e.g., security, emergency planning), which may limit the economic benefit from such actions. The regulations do not allow for restart once the licensee provides certification of the permanent removal of fuel and permanent shutdown of the plant in accordance with 10 CFR 50.82.

QUESTION 2 [addressed to Douglas Broaddus, NRC]: 10 CFR 50.54(q) allows licensees to make changes to the emergency plan provided there is no reduction in effectiveness. Once the 50.82 letter is submitted the reactor can no longer be loaded with fuel such that many accidents cannot happen. How is the removal of some positions that are no longer required a reduction in effectiveness at that plant?

ANSWER 2 [answered by Chris Gratton, NRC]: A licensee is required to submit a written certification per 10 CFR 50.82(a)(ii) once fuel has been permanently removed from the reactor vessel. A licensee may make changes to plant design and operation under the authority granted by the 10 CFR 50.59 change process including certain staffing changes. However, as stated in 10 CFR 50.59(c)(4), that change process does not apply when the applicable regulations establish more specific criteria for accomplishing changes, such as 10 CFR 50.54(q). In addition, licenses may not use the 10 CFR 50.59 change process to remove staffing positions designated in the license without prior NRC approval to amend the license. If a licensee's 10 CFR 50.54(q)(3) analyses assume that the 10 CFR 50.59 change process modified the approved emergency plan's licensing basis, then the 10 CFR 50.54(q)(3) evaluation that relied upon this rationale would be defective. As such, it would be inappropriate for a licensee to base a change to its emergency plan upon a 10 CFR 50.59 change that concludes design basis accidents and transients postulated to occur during reactor operation were no longer possible. Instead, a licensee is required to assess the proposed emergency plan changes against the most-recently NRC-approved emergency plan in determining whether prior NRC approval was required.

QUESTION 3 [addressed to Douglas Broaddus, NRC]: Does 10 CFR 50.54(hh) apply to a plant that has submitted the 10 CFR 50.82(a) certification letters?

ANSWER 3 [answered by Chris Gratton, NRC]: The plain language of 10 CFR 50.54(hh)(3) exempts all facilities that have submitted cessation of operation certifications without regard to whether there is still fuel onsite that is not in an ISFSI – i.e. fuel in the pool. This is inconsistent with the intent the Commission had when issuing the Power Reactor Security Requirements final rule. Given that NRC intended that the 10 CFR 50.54(hh)(2) requirements should apply to decommissioning facilities with fuel still in the spent fuel pool, rulemaking is being considered to correct the oversight. In the interim, for licensees where spent fuel is still onsite and not in an ISFSI, the staff may find cause to deny amendment requests that propose removing equipment, staffing, and procedural requirements from the license needed for compliance with 10 CFR 50.54(hh).

QUESTION 4 [addressed to Robert Orlikowski, NRC]: What transition does a plant in SAFSTOR go through for the groundwater protection initiative and related monitoring of potential releases from spent fuel pool or underground lines?

ANSWER 4 [answered by Robert Orlikowski, NRC]: The NRC reviews and inspects the groundwater monitoring programs at decommissioned sites that have spent fuel pools. Inspection Manual Chapter 2561, "Decommissioning Power Reactor Inspection Program," lists the core inspection procedures to be conducted at a decommissioned power reactor site. Inspection Procedure is IP 84750, "Radioactive Waste Treatment, and Effluent and Environmental Monitoring," is a core procedure that is used by inspectors to ensure that licensees effectively control, monitor, and quantify any releases as well as ensure that radiological environmental monitoring programs are effectively implemented.

QUESTION 5 [addressed to Ralph Andersen, NEI]: Is the NEI guidance intended for the reactors currently in the decommissioning transition process or for future plants that shut down?

ANSWER 5 [answered by Ralph Andersen, NEI]: The planned industry guidance for decommissioning transition will be developed by NEI from the experience and lessons-learned of the plants currently in the decommissioning transition process. The industry guidance is expected to support plants that enter the decommissioning transition process in the future. By working together and interacting with the NRC on a collective basis, the plants currently in the decommissioning transition process will obtain improvement in efficiencies and will help develop the industry guidance.

TH27 Safety Critical Software–International Perspectives

Session Chair: Steven Arndt, Senior Technical Advisor for Digital I&C, Division of Engineering, NRR/NRC, 301-415-6502, Steven.Arndt@nrc.gov

Session Coordinator: Michael Waterman, Senior Instrumentation and Controls Engineer, Division of Engineering, RES/NRC, 301-251-7451, Michael.Waterman@nrc.gov

The questions below were not answered during the above session.

QUESTION 1 [addressed to Mark Bowell, ONR, UK]: Do you perform detailed code reviews to achieve an acceptable level of confidence? Do you control approval of digital I&C platform changes and revisions? How do you exercise configuration management /version control?

ANSWER 1 [answered by Mark Bowell, ONR, UK]: ONR might expect the licensee to present evidence from independent detailed code reviews as part of their safety demonstration, depending on the safety class of the system, the reliability claim and the strength of other evidence.

ONR assess digital I&C platform changes and revisions in the context of a safety demonstration from a licensee. If a licensee changes or revises the platform used in a system important to safety, we expect them to provide an appropriate safety demonstration for the modification. The depth of the demonstration should be proportionate to the potential of the modification to compromise safety if poorly executed.

QUESTION 2 [addressed to Mark Bowell, ONR, UK]: In addition to software safety plans, are you reviewing and/or considering other I&C safety plans? What standard do you use for review of safety plans?

ANSWER 2 [answered by Mark Bowell, ONR, UK]: Software is just one constituent of an I&C system. It is unrealistic for the licensee to produce an adequate safety demonstration without planning what evidence is necessary and how this will be produced, for the I&C system as a whole.

In general, ONR's assessment is of the safety demonstration (rather than the safety plan). However, we will often meet with the licensee well before this demonstration is complete, to ensure our expectations are aligned and the demonstration is likely to be adequate in scope and depth. Plans produced by the licensee are relevant in this context.

Safety plans are covered in the introduction and chapter 1.1 of the regulator task force on safety critical software (TF SCS) common position document.

More general information on ONR's expectations for a safety demonstration is given in our technical assessment guide 51 (see http://www.onr.org.uk/operational/tech_asst_guides/index.htm).

QUESTION 3 [addressed to Mark Bowell, ONR, UK]: In the UK, have you encountered incidents or operating experience that have so reduced the reliability aspect of the safety case so as to significantly impact or cause the loss of the regulator's confidence in the safety of the system? How have you dealt with such emergent information and experience?

ANSWER 3 [answered by Mark Bowell, ONR, UK]: Yes. ONR expects licensees to be proactive in their awareness of incidents and operating experience, to consider the implication of this information on their own systems, and to modify their system and/or procedures, and its safety demonstration, to adequately compensate for relevant implications.

QUESTION 4 [addressed to Mark Bowell, ONR, UK]: What is the easiest method to become aware and informed (or even involved) in the UK's approach to digital I&C for new nuclear power plants? What is the link to ONR, MDEP technical working group, and the Task Force websites on these issues?

ANSWER 4 [answered by Mark Bowell, ONR, UK]: For ONR's generic design assessment of new nuclear plants, see http://www.onr.org.uk/new-reactors/index.htm.

For MDEP, see http://www.oecd-nea.org/mdep/

The TF SCS currently does not have a public website. The common position document is published individually by each TF SCS member. For example, ONR provides a copy at http://www.onr.org.uk/software.pdf. A 2014 revision is being prepared for publication and should be available within a couple of months.

The TF SCS is restricted to nuclear regulators and their authorized technical support organizations. If you belong to either of these and would like to get involved, please contact me.

QUESTION 5 [addressed to Mark Bowell, ONR, UK]: Could you please expand on the level of detail you as a regulator would expect to review for a Commercial-Off-the-Shelf (COTS) reactor safety system?

ANSWER 5 [answered by Mark Bowell, ONR, UK]: The TF SCS common position document uses the term "pre-existing software" to include COTS. Chapter 1.4 provides more detail.

QUESTION 6 [addressed to Mark Bowell, ONR, UK]: When assessing commercial software for a critical application, how does regulator review the utility assessment of the software developer? Does the regulatory review include reviewing user feedback and how the vendor uses this information?

ANSWER 6 [answered by Mark Bowell, ONR, UK]: ONR's assessment will focus on the adequacy of the actual system important to safety, as presented in the licensee's safety demonstration. For safety systems, this demonstration will contain evidence on the development process, including how operational experience is used to make modifications. See also question 3.

QUESTION 7 [addressed to Mark Bowell, ONR, UK]: Has the commercial grade dedication of safety critical software been considered by the Task Force explicitly?

ANSWER 7 [answered by Mark Bowell, ONR, UK]: No. The common position document is primarily aimed at safety systems. However, chapter 1.11 gives some initial consideration to how these requirements might be relaxed for systems with less safety criticality.

QUESTION 8 [addressed to Mark Bowell, ONR, UK]: Could you provide a real world example of tool validation for:

- Safety-critical software
- Significant or important to safety software
- Non-Safety Related (BOP) industrial grade software.

ANSWER 8 [answered by Mark Bowell, ONR, UK]: Real world examples of the application of tools and their validation are usually commercially protected; hence this question is best addressed via more formal exchanges such as MDEP or bi-laterals (eg US NRC / ONR).

QUESTION 9 [addressed to Mark Bowell, ONR, UK]: What is the expected QA qualification for software tools? Do they need to meet the same requirements as safety critical systems? If not, why not?

ANSWER 9 [answered by Mark Bowell, ONR, UK]: Chapter 1.5 of the TF SCS common position document covers software tools, including rationale and issues involved. The 2014 revision includes the following requirement: 1.5.3.2, The qualification requirements for an individual tool will be a function of its impact on the target software, and of how other tools or processes may or may not detect faults introduced by that tool. The combined set of tools shall be qualified to a level that preserves the required properties of the target software.

QUESTION 10 [addressed to Mark Bowell, ONR, UK]: What are the viable approaches to address common cause failures / common mode failures in an integrated digital control system for a safety-critical or safety-related solution that does not use HW/SW platform diversity?

ANSWER 10 [answered by Mark Bowell, ONR, UK]: Chapter 1.12 of the TF SCS common position document covers software design diversity for protecting against common cause failures. Platform diversity is included but the scope is much wider.

QUESTION 11 [addressed to Mark Bowell, ONR, UK]: What are the issues, if any, with implementing multiple safety-critical software modules in one integrated system on digital hardware or equipment (for example Teleperm XP or Tricon)?

ANSWER 11 [answered by Mark Bowell, ONR, UK]: ONR has not assessed either the Tricon or Teleperm XP. Our recent assessments have focused on platforms such as Teleperm XS, Common Q (both safety class 1 role) and the SPPA T2000 (S7), the latter in a lower safety class role. ONR's use of standards such as IEC 60880 (Class 1) and IEC 62138 (Classes 2 and 3) have proven to be effective for a wide range of software structures and approaches to modularization for the above platforms.

TH28 Severe Accident Codes and Analysis Applications and Fukushima Response Activities

Session Chair: Richard Lee, Branch Chief, Division of Systems Analysis, RES/NRC, 301-251-7526, Richard.Lee@nrc.gov

Session Coordinator: Matthew Humberstone, Reactor Systems Engineer, Division of Systems Analysis, RES/NRC, 301-251-7909, Matthew.Humberstone@nrc.gov

The questions below were not answered during the above session.

QUESTION 1 [addressed to Nathan Bixler, Sandia National Laboratories]: Is it possible to use MACCS as a "crisis tool" which can help making decisions for population protection?

ANSWER 1 [answered by Nathan Bixler, Sandia National Laboratories]: MACCS could be used as an emergency response tool but RASCAL, the NRC incident response tool, has several advantages for guiding emergency response:

- It models real-time weather using data from the nearby weather towers.
- It allows a rapid estimation of source term for a specific plant that can be used to estimate doses.

On the other hand, MACCS could be run to estimate pros and cons of several sheltering/evacuation strategies. RASCAL does not explicitly account for emergency response. However, this can be done (and has been done) in advance to provide risk informed guidance on protective actions.

QUESTION 2 [addressed to Randy Gauntt, SNL]: For hydrogen risk in reactor building, it seems, based on the presentation, that venting early may be detrimental. Is that correct? Can you comment on this?

ANSWER 2 [answered by Randy Gauntt, SNL]: Containment venting can both protect the containment from failure and uncontrolled releases – venting also allows the direction of H2 to the stack as opposed to an uncontrolled release into the reactor building.

QUESTION 3 [addressed to Kenji Tateiwa, TEPCO]: For KK what are FCV pipe diameters? How is hydrogen detonation in vent addressed?

ANSWER 3 [answered by Kenji Tateiwa, TEPCO]: The FCV pipe diameter is not public information. Nitrogen is used to inert the piping to prevent hydrogen detonation.

QUESTION 4: Is it time for a Regulatory Guide on how to apply MACCS?

ANSWER 4: User Guide (instead of Regulatory Guide) provides guidance on the use of MACCS.

QUESTION 5 [addressed to Randy Gauntt, SNL]: MELCOR predicts H2 will burn when a burnable concentration is reached. The energy visually observed at Fukushima 1, 3, and 4 seems more consistent with a delay in such burn until a higher concentration is reached. Please comment on the timing of the burn at these Fukushima reactors compared to the MELCOR assumed timing.

ANSWER 5 [answered by Randy Gauntt, SNL]: With respect to burning – we are not predictive in terms of burn initiators and this is handled parametrically once flammable conditions are attained. That said, we do predict inert conditions, which we think explains the delay in H2 combustion in 1F1 and the explosion was observed after we estimated flammable conditions were developing. Finally while over 700kg H2 is believed to have leaked into the refueling bay – only about 200kg is believed to have been resident in the building at any point in time during the more than 10 hours prior to the explosion.

TH29 Small Modular Reactor Licensing–Transition from Concept to Implementation

Session Chair: Anna Bradford, Branch Chief, Division of Advanced Reactors and Rulemaking, NRO/NRC, 301-415-1560, Anna.Bradford@nrc.gov

Session Coordinator: Wesley Held, Project Manager, Division of Advanced Reactors and Rulemaking, NRO/NRC, 301-415-1583, Wesley.Held@nrc.gov

The questions below were not answered during the above session.

Question 1 [addressed to Rebecca Smith-Kevern, DOE and Stewart Magruder, NRC]: What activity is there in your agency around Gen IV SMRs?

Answer 1a [answered by Stewart Magruder, NRC]: The NRC is participating in international efforts related to developing standards for non-light water reactors. For example, we recently reviewed and commented on Safety Design Criteria for sodium fast reactors that were developed by the Generation IV Forum. The NRC and the Department of Energy have been working on a joint initiative to consider what revisions may be needed to the General Design Criteria in 10 CFR 50, Appendix A, if an application for a non-light water design was submitted to the NRC for review.

Answer 1b [answered by Rebecca Smith-Kevern, DOE]: The Gen IVR&D subprogram supports the development of innovative SMR designs that may offer improved safety, functionality and affordability, and build upon existing nuclear technology and operating experience. The program supports laboratory, university and industry projects to conduct nuclear technology R&D, including the development of codes and standards, novel sensors, control systems for multiple units, and other technologies that are unique and would be useful to support development of advanced SMR concepts for use in the mid-to long-term. Emphasis is on advanced reactor technologies to support advanced small reactors that offer simplified operation and maintenance for distributed power applications, more efficient energy conversion and increased proliferation resistance and security.

Question 2 [addressed to Stewart Magruder, NRC]: Do you interact with China National Nuclear Safety Administration?

Answer 2 [answered by Stewart Magruder, NRC]: The NRC has formal bilateral and multilateral interactions with China's NNSA. For the Office of New Reactors, these interactions are focused on the Westinghouse AP1000 reactor design that is being constructed in China and the United States. In particular, there have inspector exchanges, bilateral exchanges on AP1000 experience, and sharing design review experience via the Multinational Design Evaluation Program AP1000 Working Group.

Question 3 [addressed to Stewart Magruder, NRC]: Will you require SMR licensees to collect experience with the first prototypes before installing more units?

Answer 3 [answered by Stewart Magruder, NRC]: The NRC has a robust operating experience program that will look for insights from the first SMR plants. In addition, the Institute for Nuclear Power Operations has initiatives that are designed to foster and improve plant performance industry-wide. This includes formal reviews of operating experience.

Question 4 [addressed to Stewart Magruder, NRC]: Given recent experience with design changes occurring during construction of AP1000 units at Vogtle and Summer, is design certification for the first of a kind SMR the best licensing strategy? What about prototype, standard design approval, etc.?

Answer 4 [answered by Stewart Magruder, NRC]: All of the licensing processes for new reactor designs described in 10 CFR 50 and 52 are available to SMR designers. The NRC does not have an opinion on which process is best suited to the objectives of a particular applicant.

Question 5 [addressed to NEI]: What are the challenges with using the Part 50 licensing process (construction permit) for initial SMR installations?

Answer 5 [answered by NEI]: A Part 50 CP application for an SMR would be the first such new application received by NRC in decades, after retooling much of the licensing review process in recent years for Part 52 ESP, design certification and COL applications. It may be a challenge to determine the necessary and sufficient level of detail for new CP applications.

Question 6 [addressed to NEI]: Will NEI be looking at revising aircraft impact assessments and loss of large area guidance for SMRs?

Answer 6 [answered by NEI]: Yes. Because some aspects of existing guidance for performing aircraft impact assessments for new plants is not applicable to SMR designs, NEI expects to work with applicants and the NRC staff to adjust AIA methodology as appropriate for SMRs, perhaps via an update or supplement to NEI 07-13.

Question 7 [addressed to Andy Kugler, NRC]: To what extent does ISG-027 consider different environmental impact issues and how they may need to be evaluated for SMRs deployed in remote locations (e.g. extreme hot/cold climates), such as analysis techniques and tools that differ than those used for the existing fleet?

Answer 7 [answered by Andy Kugler, NRC]: The NRC staff does not believe that its current guidance requires modification to address extreme climates at this time. There are two reasons for this. First, the current guidance focuses on the impacts to environmental resources. Regardless of location or climate, the staff will evaluate the impacts to those resources. The staff expects that the current guidance will be adequate regardless of location. To the extent the commenter may have been concerned about how an extreme climate might impact the safety of the plant, such issues are outside the scope of the environmental review and ISG-027.

Second, there has been no indication at this time of any plans to locate an SMR in a location with an extreme environment. It wouldn't be a good investment of limited staff resources to try to develop guidance for a theoretical scenario that at this time does not appear likely. At some future time, the staff might become aware of an application in an extreme environment through early engagement with a potential applicant in accordance with 10 CFR 51.40, Consultation with the NRC Staff. If the need arises for specialized guidance, then the staff will develop the necessary guidance for use on that application.

Question 8 [addressed to Rebecca Smith-Kevern, DOE]: The administration promotes "all of the above" energy sources. Furthermore, you described aggressive development of SMRs (~40/year). Yet, the same administration has no waste disposal policy. How are these positions to be reconciled?

Answer 8 [answered by Rebecca Smith-Kevern, DOE]: Nuclear energy has played an important role in avoiding carbon pollution and providing affordable energy, providing nearly 20 percent of electrical generation in the U.S. over the past two decades and representing about 60 percent of carbon-free electricity in the United States. The Department of Energy is committed to sustaining nuclear energy's role in America's low-carbon future.

Part of that commitment is offering solutions for the disposition of used nuclear fuel and high-level radioactive waste based upon the recommendations of the Blue Ribbon Commission for America's Nuclear Future (BRC). A core recommendation of the BRC was implementation of a consent-based process as a prerequisite for long-term success.

When President Obama took office, the timeline for opening Yucca Mountain had already been pushed back by two decades, stalled by public protest and legal opposition, with no end in sight. It was clear that the stalemate could continue indefinitely. The Administration made our position clear -- Yucca Mountain is not a workable solution, and we can do better.

President Obama directed Secretary Chu to establish the BRC in 2010 to conduct a comprehensive review of policies for managing the back end of the nuclear fuel cycle and to provide recommendations for developing a safe, long-term solution to managing the nation's used nuclear fuel and nuclear waste.

In January 2013, the Administration issued the Strategy for the Management and Disposal of Used Nuclear Fuel and High-Level Radioactive Waste. The strategy embraced the core findings of the BRC and affirmed that any workable solution for the final disposition of used fuel and nuclear waste must be based not only on sound science, but also on achieving public acceptance at the local, and state and tribal levels.

The Administration supports working with Congress to develop a consent-based process that is transparent, adaptive and technically sound. The BRC emphasized that flexibility, patience, responsiveness and a heavy emphasis on consultation and cooperation with all parties are necessary in the siting process and in all aspects of implementation.

The BRC and the Administration strategy also emphasize the importance of pursuing consolidated interim storage in parallel with looking at alternative sites for geologic disposal. A first priority is consent-based siting of a pilot-scale storage facility to accommodate used fuel from shutdown reactors. This is something we should all come together to support as a sensible first step towards building a robust used fuel disposal system that will stand the test of time.

Question 9 [addressed to Rebecca Smith-Kevern, DOE]: Would DOE consider first-of-a-kind support to US-based vendors to enter non-US jurisdictions?

Answer 9 [answered by Rebecca Smith-Kevern, DOE]: The Department of Energy (DOE) sees a strong opportunity for American leadership in small modular reactor development. Nuclear energy is vital to building a more sustainable, low-carbon energy future, and DOE is committed to supporting a strong domestic nuclear energy industry that can lead an increasingly competitive and growing global market – manufacturing these technologies here and exporting them around the world.

TH30 The Future of the NRC Force-on-Force Inspection Program

Session Chair: Patricia Holahan, Director, Division of Security Operations, NSIR/NRC, 301-287-3664, Patricia.Holahan@nrc.gov

Session Coordinator: Ron Albert, Branch Chief, Division of Security Operations, NSIR/NRC, 301-287-3661, Ronald.Albert@nrc.gov

Questions submitted during the above session were answered during the session's Q/A period.

TECHNICAL SESSIONS
Thursday, March 13, 2014, 3:30 p.m. – 5:00 p.m.

TH31 Future Vision of Spent Fuel Storage–Back-End Friendly Fuel Designs and Holistic Safety Security Interface

Session Chair: Mark Lombard, Director, Division of Spent Fuel Storage and Transportation, NMSS/NRC, 301-287-0673, Mark.Lombard@nrc.gov

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Questions submitted during the above session were answered during the session's Q/A period.

TH32 New and Expanding Nuclear Energy Programs: Experience Confronting Regulatory Challenges

Session Chair: Nader Mamish, Office Director, OIP/NRC, 301-415-1780, Nader.Mamish@nrc.gov

Session Coordinator: Danielle Emche, International Relations, OIP/NRC, 301-415-2644, Danielle.Emche@nrc.gov

Questions submitted during the above session were answered during the session's Q/A period.

TH33 New Reactor Construction Successes and Challenges

Session Chair: Andrea Valentin, Deputy Director, Division of Construction Inspection and Operational Programs, NRO/NRC, 301-415-3287, Andrea.Valentin@nrc.gov

Session Coordinator: Carl Weber, Reactor Operations Engineer, Division of Construction Inspection and Operational Programs, NRO/NRC, 301-415-6689, Carl.Weber@nrc.gov

The questions below were not answered during the above session.

QUESTION 1 [addressed to Panel]: How are existing projects in the US able to gain access to China's extensive construction operating experience? Comment on transparency and international sharing.

ANSWER 1 [answered by Alan Torres, SCE&G]: First, remember that the only part that is considered standard plant is the nuclear island. We gain excellent Operating Experience from Westinghouse on the China plants. We have also set up areas like training and operations that share openly.

QUESTION 2 [addressed to Rahsean Jackson, NRC and Alan Torres, SCE&G]: Are multiple CAP programs on site still causing confusion? Have you considered all parties inputting to one CAP program?

ANSWER 2a [answered by Rahsean Jackson, NRC]: Multiple CAPs don't cause confusion; it simply requires more diligence and discipline by the inspectors. Typically, the NRC has full access to a licensee's CAP database which makes it relatively easy to track specific issues through corrective action program process. Our current situation requires the inspectors to track and follow one issue in multiple programs without direct access to each program database. In some cases the inspectors have to request a hard copy of a CAP document at different stages of the corrective action process to verify the issue was properly assigned and dispositioned by the responsible organization.

ANSWER 2b [answered by Alan Torres, SCE&G]: We have evaluated combining the CAP programs, but that presents numerous Information Technology issues. We are able to coordinate all the different programs by using a charter that provides direction for interface of the programs.

QUESTION 3 [addressed to Rahsean Jackson, NRC]: Have you gotten sufficient support from NRC headquarters staff? What do you think the most urgent need is at this point?

ANSWER 3 [answered by Rahsean Jackson, NRC]: Yes, the residents are currently getting ample support from HQ in a variety of ways. We have the Technical Assistance Request (or TAR) process which we use to ask questions which are technical or regulatory in nature. We have a Senior Project Manager for VC Summer who processes these TARs and supports all HQ program responsibilities. We also have a licensing Project Manager who is responsible for processing license amendments submitted by SCE&G and licensing related inquiries from the residents.

TH34 Radiation Protection Regulations and Computer Codes

Session Chair: Rebecca Tadesse, Branch Chief, Division of Systems Analysis, RES/NRC, 301-251-7490, Rebecca.Tadesse@nrc.gov

Session Coordinator: Gladys Figueroa, Reactor Systems Engineer, Division of Systems Analysis, RES/NRC, 301-251-7545, Gladys.Figueroa@nrc.gov

Questions submitted during the above session were answered during the session's Q/A period.

TH35 Reactor Oversight Process Enhancements

Session Chair: Allen Howe, Deputy Director, Division of Inspection and Regional Support, NRR/NRC, 301-415-1004, Allen.Howe@nrc.gov

Session Coordinator: Gabriel Levasseur, Reactor Operations Engineer, Division of Inspection and Regional Support, NRR/NRC, 301-415-1487, Gabriel.Levasseur@nrc.gov

The questions below were not answered during the above session.

QUESTION 1 [addressed to Rani Franovich, NRC]: Is the NRC developing a mobile application to provide ROP related information to stakeholders?

ANSWER 1 [answered by Rani Franovich, NRC]: A mobile application to convey ROP information to stakeholders has been discussed. However, NRC has decided to focus its resources on making its website and information (such as ROP data) more mobile-friendly and open, as a cost effective alternative to developing an APP. As part of the ROP Enhancement Project the NRC will continue to explore ways to enhance communications with stakeholders.

<u>TH36 Safety Concern: Degradation of Neutron Absorbing Materials in the Spent Fuel Pool</u>

Session Chair: Christopher Jackson, Branch Chief, Division of Safety Systems, NRR/NRC, 301-415-3456, Christopher.Jackson@nrc.gov

Session Coordinator: Jocelyn Lian, Technical Assistant, Division of Engineering, NRR/NRC, 301-415-4666, Jocelyn.Lian@nrc.gov

Questions submitted during the above session were answered during the session's Q/A period.