U.S.NRC United States Nuclear Regulatory Commission *Protecting People and the Environment*

THE 24TH ANNUAL REGULATORY INFORMATION CONFERENCE RIC 2012 March 13–15

ANSWERS TO UNANSWERED QUESTIONS FOR RIC 2012

Tuesday, March 13, 2012

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OPENING SESSION
Tuesday, March 13, 2012, 9:30 a.m. – 11:30 a.m.

EDO Remarks - "An Overview of NRC Operations" - Bill Borchardt, Executive Director for Operations

The questions below were not answered during the above session:

Question 1: The restrictions on foreign ownership seem to have been interpreted as more restrictive than in the past. Is there now more conservative criteria established for staff guidance?

Answer 1: No, the NRC's criteria for addressing foreign ownership have not changed. The NRC provides guidance on foreign ownership in the "Final Standard Review Plan on Foreign Ownership, Control or Domination," (64 FR 52355, September 28, 1999). The guidance has not been revised. The intent of the guidance is to provide broadly applicable criteria to address the "perhaps limitless creativity involved in formulating corporate structures and arrangements" on a case-by-case basis. The guidance also provides a multitude of methods that may be tailored to fit a wide variety of foreign investments in domestic nuclear power projects, to the extent allowed by the Atomic Energy Act of 1954, as amended. However, there is no "safe harbor" that would eliminate the need of the NRC to review the corporate arrangements, when foreign participation is involved. The number of cases where foreign ownership, control, or domination has arisen has increased due to the globalization of economic activity; likewise, the complexity of the corporate arrangements also appears to have increased. In response, the NRC's reviews have become more numerous and detailed.

Question 2: Could you briefly discuss the quality management issues that you have observed with nuclear vendors and how the NRC's regulatory scheme will address these shortcomings, especially as new reactor construction continues?

Answer 2: Technical reviews performed by NRC staff have indicated that licensee oversight of vendor-supplied technical analysis products could be strengthened. The NRC staff continues to identify technical deficiencies in license amendment submittals in the areas of licensing basis safety analysis, as well as vendor supply issues related to design, 10 CFR Part 21, and regulatory pass down requirements (e.g., Appendix B to 10 CFR Part 50). The staff believes that additional attention is needed on the part of the industry to ensure licensee quality assurance (QA) oversight functions are being properly implemented.

Addressing new construction activities, staff from the NRC's Office of New Reactors performs routine inspections focusing primarily on vendors supporting these activities. The staff uses the Vendor Inspection Program (VIP) Plan, which establishes an overall approach, including goals, priorities, performance metrics, and resource management strategies for VIP activities. It also verifies that reactor applicants and licensees are fulfilling their regulatory obligations with respect to providing effective oversight of the supply chain through a number of activities, including performing vendor inspections that will verify the effective implementation of the vendor's QA program, establishing a strategy for vendor identification and selection criteria, and ensuring vendor inspectors obtain necessary knowledge and skills to perform inspections. In addition, the VIP addresses interactions with nuclear consensus standards organizations, industry and external stakeholders, and international constituents.

Regarding fuel thermal conductivity degradation, the NRC staff has raised concerns related to fuel vendor processes used to identify an error. These concerns were not primarily the result of a deficient licensee QA program related to a technical submittal.

Question 3: What is the staff's vision on what the real tangible impact on licensees in response to Fukushima?

Answer 3: The NRC staff has been extremely focused over the past year on (1) identifying lessons-learned from Japan's March 11, 2011, earthquake and tsunami; and (2) making any necessary enhancements to its regulatory system in a systematic and methodical manner. As such, the Commission approved the NRC staff's implementation plan for the eight Tier 1 activities outlined in SECY-11-0137, "Prioritization of Recommended Actions to be Taken in Response to Fukushima Lessons Learned," dated October 3, 2011. These Tier 1 items are being implemented without unnecessary delay and will result in safety enhancements across the U.S. nuclear fleet. The eight Tier 1 items include the following:

(1) NTTF Recommendation 2.1 – This recommendation is being addressed through a request for information to NRC licensees in accordance with 10 CFR 50.54(f), which was issued on March 12, 2012. Licensees have been asked to perform, and provide the results of, a reevaluation of the seismic and flooding hazards at their sites using present day NRC requirements and guidance.

(2) NTTF Recommendation 2.3 – This recommendation is being addressed through a request for information to NRC licensees in accordance with 10 CFR 50.54(f), which was issued on March 12, 2012. Licensees have been requested to develop a methodology and acceptance criteria for the performance of seismic and flooding walkdowns, and complete the walkdowns within 180 days of the NRC's May 2012 endorsement of the walkdown procedures, or about November 2012.

(3) NTTF Recommendation 4.1 – This rulemaking will enhance a plant's ability to respond to a station blackout (SBO) and is expected to be completed within 24-30 months. An advance notice of proposed rulemaking (ANPR) to solicit public comment on this rulemaking was issued on March 20, 2012.

(4) NTTF Recommendation 4.2 – This order, which was issued on March 12, 2012, requires licensees to develop strategies to mitigate beyond design basis natural phenomena, addressing both multi-unit events and reasonable protection of equipment identified under such strategies.

(5) NTTF Recommendation 5.1 – This order, which was issued on March 12, 2012, requires licensees with boiling water reactors with Mark I or II containments to have a reliable hardened vent to remove decay heat and maintain control of containment pressure within acceptable limits following beyond events that result in the loss of active containment heat removal capability or prolonged SBO.

(6) NTTF Recommendation 7.1 – This order, which was issued on March 12, 2012, requires licensees to have a reliable indication of the water level in spent fuel storage pools capable of supporting identification of spent fuel pool water level by trained personnel, to facilitate actions to restore water level, if necessary.

(7) NTTF Recommendation 8 – This rulemaking addresses the integration of emergency procedures and is expected to be completed in 2016. The staff expects to publish an ANPR soliciting stakeholder feedback on this rulemaking in mid-April.

(8) NTTF Recommendation 9.3 – This recommendation is being addressed through a request for information to NRC licensees in accordance with 10 CFR 50.54(f), which was issued on March 12, 2012. Licensees have been requested to assess their current communications system and equipment under conditions of onsite and offsite damage and prolonged SBO, and perform a staffing study to determine the number and qualifications of staff required to fill all necessary positions in response to a multiunit event.

Question 4: Will the new building house a new gym to accommodate the staff that's returning?

Answer 4: The new Three White Flint North (3WFN) building will not house a gym or fitness center. We are currently evaluating future fitness center needs that may result as staff relocates from the interim leased buildings to the 3WFN and Two White Flint North buildings. We will work to satisfy needs identified as a result of the evaluation.

Question 5: Since 2003, the number of NRC employees has increased from 3,000 employees to 4,000. During this time the number of reactor licenses has increased by 1 and the number of non-reactor licenses has decreased with additional Agreement States. And the Yucca Mountain program was terminated. Please describe in specific terms where the budget challenges exist?

Answer 5: The NRC experienced substantial growth from 2005 through 2010 to respond to the projected increased workload, which was predominately as a result of the renewed interest in nuclear energy and the industry's plans to build new nuclear power reactors and other nuclear facilities. From fiscal year 2005 to FY 2010, the NRC's total budget increased by 60 percent from $669 million to $1,067 billion. The NRC's projected workload has now leveled off. Although, some projected workload has increased to implement the recommendations from the Fukushima Near-Term Task Force and other changes, this growth in demand is partially offset by workload reductions associated with a slower pace of new nuclear power plant applications and construction. In addition, the Nation's mounting debt situation has resulted in renewed interest from the President, Congress and other stakeholders in reducing Federal Government spending. As a result, the NRC's budget is likely to remain flat or decline during the next five years.

To successfully ensure that NRC fulfills its safety and security mission, we must increase both our effectiveness and efficiency and to rethink the way we do business to eliminate unnecessary duplicative spending and "do better with less." Fortunately for the NRC the financial and programmatic challenges of today do not require a whole new approach to conducting business. By taking the following actions over the next several years to control the agency's operations costs we will be able to maintain a high-quality workforce with the required skills to effectively carry out the NRC's safety and security mission in this period of no growth or declining budgets.

- Implementation of innovative business practices, including the Transforming Assets into Business Solutions recommendations to enhance resource management tools, and greater emphasis on teamwork between offices.
- Implementation of the NRC's 21st Century Acquisition Program to leverage agency resource and contract dollars by using effective and efficient business processes and automated tools.
- Continue to conduct Business Process Improvement efforts to increase product quality, decrease costs, and result in improved effectiveness.
- Emphasis the importance of interdependence in planning and execution to ensure that the NRC minimizes the duplication of programs, processes and procedures
- Development and implementation of a strategic organization and position management plan that looks five years into the future for staffing, recruitment and hiring
- Further enhance agency processes and standardized information technology tools.

Question 6: With resource tightening, and fewer staff, what is your position on productivity and efficiency to compensate for reductions so that initiatives can continue?

Answer 6: The staff's long-term plan is to reduce operational costs, devote sufficient resources to license and regulate the uses of nuclear facilities and materials, while remaining adaptive and flexible to address externally driven changes and carry through on necessary new initiatives. At the NRC, both programs and corporate offices support and contribute to the agency's mission; therefore it is imperative that staff members work together. They collectively must demonstrate a commitment to greater interdependence by communicating and collaborating to enhance cross-organizational, process, program, and procedural effectiveness, as appropriate.

Question 7: What is the status of re-integration of NSIR to NRR to "normalize" the security framework with other regulatory programs?

Answer 7: The NRC has no current plans to reintegrate the NSIR office with NRR to "normalize" the security framework with the other regulatory program. However, the staff is currently working to reintegrate the Security cornerstone into the performance assessment program, governed by Inspection Manual Chapter (IMC) 0305, "Operating Reactor Assessment Program." The staff issued a Regulatory Issue Summary (RIS) on March 18, 2012, informing stakeholders of the reintegration of the Security cornerstone into the performance assessment program and the effective date of that reintegration. IMC 0305 and IMC 0320, "Operating Reactor Security Assessment Program," will be revised in late June to reflect security reintegration and will become effective July 1, 2012.

The public website will be updated in early August to include Security cornerstone inspection findings and performance indicators (PIs). However, detailed information will remain non-publicly available. The website will indicate security inspection findings and PIs that were determined to be of very low significance, i.e., Green, and use blue to indicate greater-than-Green (White, Yellow, or Red) security performance issues. No other detailed information regarding security performance will be publicly available. The regional offices will use the revised guidance in IMC 0305 to assess plant performance during the CY2012 mid-cycle assessment meetings and will issue one combined assessment letter. The integrated assessment and issuance of one assessment letter will continue from that point on.

Some stakeholders have suggested that the Security cornerstone has a disproportionate number of greater than Green findings when compared to the other cornerstones. Since the Security cornerstone is the only cornerstone in the Safeguards strategic performance area, it accounts for all of that strategic performance area's greater than Green findings. On the other hand, the Reactor Safety strategic performance area is represented by four cornerstones (Initiating Events, Mitigating Systems, Barrier Integrity, and Emergency Preparedness), and each Reactor Safety cornerstone contains only a portion of the total Reactor Safety findings, whereas the Security cornerstone contains all of the Safeguards findings. If one compares greater than Green findings by strategic performance area, the distribution of findings between the Reactor Safety and Safeguards strategic performance areas is fairly balanced.

TECHNICAL SESSIONS
Tuesday, March 13, 2012, 3:00 p.m. – 4:30 p.m.

T1 *- Approaching Closeout of Generic Safety Issue 191, Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance*

Session Chair: William Ruland, NRR

Session Coordinator: Stewart Bailey, NRR
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All questions submitted during the above session were answered during the session's Q/A period.

T2 *- Discussion of Seismic Hazard Evaluation and the Regulatory Response to Recent Seismic Events*

Session Chair: Nilesh Chokshi

Session Coordinator: Jenise Thompson, NRO
301-415-0735, Jenise.Thompson@nrc.gov

The questions below were not answered during the above session:

Question 1: When will seismic walkdown procedures be available to the utilities?

Answer 1: The NRC expects to endorse (with or without exception) the industry walkdown procedures at the end of May 2012.

Question 2: What is the current target date for the seismic walkdown personnel training sessions?

Answer 2: Based on our recent interaction with the industry the schedule developed in the recently issued 10 CFR 50.54(f) request for information letter, the training for the seismic walkdowns will begin in June 2012.

Question 3: Why is the NRC not leading a similar WUS SSHAC Level 3 for source and ground motion?

Answer 3: There are only four WUS nuclear power plant sites and each is located in a unique geologic and tectonic setting. Therefore, the staff believes that they are best handled on a case-by-case basis by the individual licensees rather than a large regional study.

Question 4: Chairman Jaczko called the Fukushima event a failure. I see it as a failure of geosciences to assess and predict the hazard. Given that our seismic hazard characterization is based on the same science, what should we do?

Answer 4: The NRC is currently requesting that operating reactor licensees and holders of construction permits under 10 CFR Part 50 reevaluate their seismic hazards using present-day seismic information and regulatory guidance. The current NRC regulatory guidance for assessing seismic hazard is based on a probabilistic approach that places an emphasis on characterizing the uncertainty associated with the size, location, and frequency of earthquakes and the associated ground motion. Therefore, rather than selecting a "maximum" design earthquake, the probabilistic approach develops a suite of different earthquake scenarios in order to capture the uncertainties in the size, location, rate of the earthquakes and associated ground motions.

Question 5: What will be the process for NRC review and approval of the seismic hazard reevaluation and GMRS?

Answer 5: The NRC staff will review the information received from licensees in response to the 10 CFR 50.54(f) letter issued on March 12, 2012 to determine if the seismic hazard has been properly characterized. This review will focus mainly on the local site response as it is expected that licensees will use established and previously approved seismic source and ground motion models. After this review, the staff will evaluate whether the licensees will need to perform a further seismic risk evaluation for their plants. Since the WUS sites will need to develop regional and local seismic source and ground motion models for their sites, the NRC staff will perform a thorough review of the seismic source and ground motion models as well as the site response characterization.

Question 6: Will NRC require additional site characterization at operating plants?

Answer 6: No. The recently issued 10 CFR 50.54(f) letter is a request for information that includes the NRC staff's acceptable approach for the seismic hazard reevaluations. As such, the NRC staff encourages the gathering of additional field data for the licensees that do not have extensive geotechnical data sets; however, the NRC is not planning to require that additional field data be collected.

Question 7: In the spirit of "the plants tell us about damage," why not let the Onogawa (Japan) plant tell us how much margin we really have before we jump into seismic PRAs and this program?

Answer 7: The Onogawa plant is located in a unique geologic and seismotectonic setting. For the nuclear power plants located in the U.S., this type of geologic setting does not exist. As such, the NRC staff is requesting that licensees characterize the seismic hazard in the region surrounding their sites to determine the need for further risk evaluation in the form of seismic PRA or margins assessment.

Question 8: Is the CEUS-SSC study likely to modify significantly the DBE of some nuclear power plants?

Answer 8: The reevaluated hazard for nuclear power plants in the CEUS will be characterized by the ground motion response spectrum (GMRS) for their sites. The new GMRS for CEUS nuclear power plants will be based on the new source models (CEUS-SSC), as well as ground motion prediction equations and an evaluation of local geologic conditions. This reevaluated hazard will then be compared to the existing seismic design basis. Licensees may then need to perform a further seismic risk evaluation for their plants if the reevaluated hazard exceeds the plant's seismic design basis. This seismic hazard and risk evaluation constitutes Phase 1 of Recommendation 2.1. During Phase 2, the NRC will determine whether changes need to be made to the plant's design bases. The DBEs for nuclear power plants are deterministic and are part of the plant's licensing basis. The probabilistic GMRS will be used for comparison purposes to determine if additional risk evaluations are necessary.

Question 9: What action will the NRC take for those plants which do not complete the seismic walkdown on schedule?

Answer 9: The NRC staff will engage any licensees that cannot complete the seismic walkdown on schedule to ensure the NRC staff receives the information it requires in a timely manner so that it can make its determination whether further regulatory action is needed. Should a licensee choose not to participate in the information collection, the NRC staff may consider other regulatory actions to obtain the information required to ensure that plant vulnerabilities are identified and that seismic protection features' monitoring and maintenance practices are deemed to be adequate.

Question 10: How many, if any, US NPPs have seismic PRAs now (beyond IPEEE)?

Answer 10: For the Individual Plant Examination of External Events (IPEEE) program, several of the licensees performed a seismic PRA (about 25). There may be additional plants that performed a seismic PRA (less than 5) but they have not been submitted to the NRC.

Question 11: For licensees and design bases established prior to App A to Part 100, where the DBE does not meet the current SRP, will licensees still be allowed to compare their new GMRS to their DBE equivalent?

Answer 11: Yes**.** Licensees will compare the new GMRS to the Safe Shutdown Earthquake ground motion or DBE equivalent spectrum for their plants.

***T3** - Domestic and International Supply Chain Oversight*

Session Chair: Michael Johnson, NRO

Session Coordinator: Cindy Rosales-Cooper, NRO
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All questions submitted during the above session were answered during the session's Q/A period.

***T4** - Emerging Issues in the Nuclear Fuel Cycle*

Session Chair: John Kinneman, NMSS

Session Coordinator: Booma Venkataraman, NMSS
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Questions submitted during the above session were answered during the session's Q/A period.

***T5** - Mandatory Hearings at the NRC – Perspectives from Participants and Decision-Makers*

Session Chair: William Froehlich, ASLBP

Session Coordinator: Sherverne Cloyd, ASLBP
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Questions submitted during the above session were answered during the session's Q/A period.

***T6** - International Panel Discussion on Assessing Environmental Risk from Radioactive Releases*

Session Chair: Thomas Nicholson, RES

Session Coordinator: Steven Schaffer, RES
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The questions below were not answered during the above session:

Question 1: A past president of the ICRP stated that Committee 5's work is only aimed at plugging a "conceptual gap" since ICRP believes that human radiation protection standards provide for adequate protection of the environment. Is this still an operative statement? If so, what is the practical use of Committee 5's work?

Answer 1: ICRP refers to the International Commission on Radiological Protection. Committee 5 of the ICRP focuses on the protection of the environment. The premise of the question is incomplete and misleading because it only addresses a portion of ICRP's position on the protection of the environment stated in ICRP Publications 91 and 103. Although the ICRP believes human radiation protection standards are likely to be protective of the environment, it also believes that it is now necessary to consider a wider range of environmental situations, irrespective of any human connection with them. The ICRP recognizes that some international requirements and a much more environmentally aware public now call for a direct and explicit demonstration that the environment is being protected. ICRP also believes that the radiation protection community needs a clearer framework in order to assess the relationships between exposure and dose, and between dose and effect, and the consequence of such effects, for non-human species.

Question 2: In light of Chairman Jaczko's statement that land contamination (such as at Fukushima Daiichi) is widely seen as unacceptable, should the United States expand the protection range of its nuclear regulations from human health and safety to a wider range of biota?

Answer 2: NRC already address radiological impacts on non-human biota in our licensing process. For example, NRC prepares an environmental impact statement (EIS) for all nuclear power facilities. These EIS include evaluations of the radiological impacts on non-human species. Many of these evaluations use quantitative techniques much like those presented in this conference session. In addition, the Near-Term Task Force made some recommendations to further enhance NRC's regulatory framework, and the Commission has tasked the staff with reviewing those recommendations.

T7 *- Risk Assessment: Emerging Challenges and Opportunities*

Session Chair: Donald Dube, NRO

Session Coordinator: Eric Powell, NRO
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The questions below were not answered during the above session:

Question 1: In the past, large break LOCAs used RCS piping. Due to the size of RCS pipes with respect to other system openings such as manways, in the SMR designs these manway openings maybe limiting for LBLOCA. Since there are numerous instances of bolting degradation (valves, reactor coolant pump flanges), please describe how PRA will reflect these changes.

Answer 1: The NRC has not received any Small Modular Reactor (SMR) design certification applications for review, so no such determination has yet been made. Based on their unique plant design, each SMR vendor will evaluate initiating events that cause an increase or decrease in reactor coolant inventory, decrease in RCS flow, etc., to establish the design basis accidents. Operating experience should also be assessed for applicability to their design and appropriately incorporated. Through this process, the significance of bolted closures may be identified in the application as a risk insight when the results of the PRA are documented.

Question 2: What model will NRC use for fire significance determination under the ROP? If it is a SPAR based on NUREG-6850, what if the licensee has a plant-specific model that is not consistent with NFPA-805 and NUREG-6850?

Answer 2: It is a goal of the NRC's Standardized Plant Analysis Risk (SPAR) risk model development and maintenance program to represent the as-built, as-operated plant to the extent needed to support operational event and condition analyses. Consistent with this approach, and to the extent practical, the NRC bases the SPAR model on the best available information. As a practical matter, a significant motivation for the development of fire PRAs has been availability of the performance-based fire protection alternative described in NFPA-805. Since the NUREG/CR-6850 reflects the current state-of-practice for NFPA-805 fire PRA development, recent fire SPAR model development efforts have focused on PRAs developed in a manner consistent with NFPA-805 and NUREG/CR-6850. For plants that do not have an NFPA-805 compliant fire PRA under the voluntary 50.48(c) initiative, the NRC uses a number of tools and guidance available to estimate the appropriate risk contribution with respect to ROP metrics (i.e., RASP Handbook, NRC Inspection Manual Chapter 0609, Appendix F). While the accuracy in these estimates will be limited by information availability, they have been used successfully in ROP activities. The use of fire PRA information compliant with NFPA-805 provides an additional level of confidence in the estimation of risk metrics for NRC's internal purposes.

Question 3: Please address how shutdown PRA addresses the use of temporary fuel storage baskets located in refueling cavities of existing reactors and proposed reactors. Please address how this relates to MODE 6 definition applicability only while fuel is in the reactor vessel, without mention of the refueling cavity storage locations.

Answer 3: We are not aware of any domestic reactor shutdown PRAs that address spent fuel in the reactor cavity. The risk from spent fuel in these locations would typically be handled via a spent fuel pool PRA, if deemed necessary. Few utilities have conducted PRAs of their spent fuel pools, and there is no regulatory requirement that would prompt this. The NRC has conducted two past spent fuel pool PRAs for specific purposes (NUREG-1353 related to resolving a generic issue of beyond-design-basis accidents in spent fuel pools; NUREG-1738 related to decommissioning). An additional reactor and spent fuel PRA is under development, as described in SECY-11-0089. In general, this type of analysis is not performed in the context of a PRA because it involves a limited number of fuel bundles stored in this condition for a very limited period of time, and is therefore likely to be a comparatively small contributor to the overall risk.

Question 4: PRA standards have outstripped licensee's ability to adopt these standards, substantially restricting licensee's ability to use PRA in licensing actions. Is there any plan to take a break on new standards and allow licensees to catch up?

Answer 4: The NRC allows a year for the licensees to upgrade their PRA to meet the standard after it is endorsed if the licensee elects to use the PRA in a risk-informed application. Moreover, it takes the NRC about a year to endorse a new or revised standard which effectively gives the licensee about two years to upgrade their PRA. In addition, the NRC has prioritized the need for new standards to be commensurate with the schedule for risk-informed applications for both operating and new reactors. For example, the NRC has placed a very low priority on the development of new standard such as the PRA standard for non-light-water reactors.

Question 5: Are all SPAR models available to the public? How does one get them?

Answer 5: As discussed in SECY 2004-0191, "Withholding Sensitive Unclassified Information Concerning Nuclear Power Reactors from Public Disclosure," detailed computer risk models (such as the Standardized Plant Analysis Risk model) are not released to the public.

Question 6: How is uncertainty considered in each of these categories: (a) DB risk (b) enhancements risk (c) residual risk?

Answer 6: Uncertainties would need to be considered when making decisions about appropriate barriers, controls, and operator actions to address a particular scenario. In the design basis events, this is often handled through safety margins and other conservatisms built into the definition and analysis of the scenario. The inclusion and disposition of design enhancement scenarios would likewise need to consider the uncertainties associated with initiating events, plant response, etc., in determining what barriers, controls, and operator actions should be established. Factors such as conditional failure probabilities of barriers might be appropriate to consider for identifying and resolving cliff-edge effects. If the Risk Management Task Force (RMTF) recommendations are pursued, the handling of uncertainties is an area that would be defined further during interactions with stakeholders and the development of specific rules and guidance.

Question 7: With regard to the proposed risk management regulatory framework, what would be your position when considering the concept of “practically eliminated” situations as proposed by European regulators for new NPPs with the purpose of having no circumstances requiring evaluation?

Answer 7: There are several possible alternative approaches within the broader framework proposed by the RMTF. One approach would define thresholds for inclusion and requirements to implement barriers, controls, and operator actions that represent risks (as measured by core damage frequency or other metrics) below defined acceptance criteria. Another approach would likewise define thresholds for inclusion for events but would require cost/benefit analyses to identify and implement cost-effective enhancements. In both of these approaches, the criteria would likely include measures such as core damage frequency, large release frequency, or other characterizations of risk. A comparison to “practically eliminated” would therefore be difficult unless the subjective terms were somehow equated to numerical risk measures.

Question 8: In the new approach of defense in depth, how do you integrate the impact of human factors and risk of inappropriate human behavior?

Answer 8: It would be desirable to integrate human factors as both contributors to risk as well as recognizing the role of personnel in preventing, mitigating, and containing the possible release of radioactive materials as a result of transients or accidents. In attempting to address human reliability within the analyses performed to address an issue, the decision makers will need to consider the limitations and uncertainties within those supporting analyses and then make judgments on the appropriate mix of engineered safeguards and procedural controls.

Question 9: Have you considered near term rulemaking to allow implementation as an option for future applicants without having to apply for an exemption(s) from prescriptive rules such as 50.46?

Answer 9: If the RMTF recommendations are pursed, it is likely that rulemaking would be required within the power reactor program. In addition to the specific addition of regulations needed to implement the design enhancement category, it may be possible to identify existing regulations, such as 10 CFR 50.46, that could be revised to take advantage of the new levels of controls added to beyond-design-basis accident scenarios. An example could be that breaks beyond a transition break size would be considered within the enhancement category while breaks below the transition break size would remain design basis events.

Question 10: New reactor designs require features to mitigate against severe accident challenges (10 CFR 52.47(a)(23)). How would the risk-informed performance-based defense in depth risk management framework accommodate this approach?

Answer 10: There are several possible alternatives within the broader RMTF proposed framework but the existing severe accident features within new plant designs could be modeled and credited within the analysis of scenarios. These existing features could, for example, result in a scenario being identified within the residual risk category instead of requiring additional analysis as a design enhancement scenario to determine if there are additional barriers, controls, or operator actions that could identified and implemented to address the subject scenario.

Question 11: Since the evolutionary long term goal is a risk-informed regulatory process (for all applications), what is the current and medium-term plan for educating and training all staff on PRA analysis and related decision-making?

Answer 11: In 2006, the NRC Office of Nuclear Reactor Regulation began qualification and training programs for technical review, inspection, and project staff. In 2007, as part of infrastructure development for the newly formed office, the Office of New Reactors developed a similar program. Part of the general qualification (for all technical staff) includes understanding the use of risk information in regulatory activities. To fulfill this qualification activity, the staff member could either attend a course, “Risk-Informed Regulation for Technical Staff,” or review specific policy statements, fact sheets, and other documents to gain an understanding of the practical application of risk in their areas of responsibility. To be certified, a technical reviewer must then demonstrate understanding of the NRC’s risk-informed approach to regulation. Although the NRC does not plan to make risk analysts of every member of the technical staff, there are PRA related classes for PRA professionals (i.e., technical reviewers, researchers, senior reactor analysts), as well as basic courses on PRA so that technical managers, inspectors, etc., can acquire an appropriate understanding of the technology. A pilot course on the adequacy of models for risk-informed decisions has also been developed, which includes training on accurately communicating risk information to decision-makers. In addition to NRC internal training, there are excellent training courses, seminars, and conferences offered by a variety of resources that members of the NRC staff are encouraged to attend.

Question 12: Risk-informed surveillance times - Utilities are embracing on-line monitoring systems, continuous monitoring, wireless equipment surveillance. Can utilities get “credit” for using OLM of I&C systems in PRA/risk-informed space?

Answer 12: The staff is not aware of an application in which a licensee has proposed to credit on-line or continuous monitoring systems or wireless equipment surveillances to justify an extension to a surveillance frequency using risk arguments. The staff also is not aware of how, or to what extent, such systems might be credited within the plant-specific PRA to justify such an extension to a surveillance frequency. Currently there are some components that have continuous monitoring, but this monitoring is not used to change a surveillance frequency, but rather is typically provided because an actual surveillance for that component should not be performed at full power conditions. As an example, in boiling water reactors (BWRs) squib/explosive valves are used in the standby liquid control system. Since the actuation of these valves is destructive of the valve internal mechanism, they have continuous monitoring of electrical current to the valves. However, a valve is still tested (and then replaced) every outage to ensure the valve would have performed its safety function.

Since such credit has not been proposed to justify a risk-informed extension to a surveillance frequency, the NRC staff has not developed a position on the use and viability of the crediting on-line or continuous monitoring systems in the context of risk-informing surveillance test frequencies.

Question 13: Do you believe there is a sound conceptual basis for the notion that decision margins can offset blackswan/unknown unknown events in PRA? If so, please elaborate.

Answer 13: Margin is one tool to ensure adequate safety, but not the only tool. It is clear that margin offsets some unknowns. It can never be certain that a given margin offsets all unknowns. That's one reason the NRC approach to regulation is not based on risk alone. Conservative, deterministic methods complement the use of PRA and safety margins.

Question 14: NRC licensed 100 plants with less than a 1000 staff. NRC now has about 3000 staff and only one plant has a new COL. Is overhead excessive?

Answer 14: Today, the Office of New Reactors has fewer than 450 employees engaged in the pre-application, certification, or recertification of six different new reactor designs and several advanced reactor designs. The office also evaluates the safety and environmental impact of proposed power plants. Other NRC staff support the current nuclear plant operating reactors, as well as test reactors, nuclear materials, fuel cycle facilities, etc.

These new and advanced reactor designs may use a new licensing process that relies on standardization as specified in Title 10 of the *Code of Federal Regulations*, Part 52 (10 CFR 52). Once a design has been certified, the applicant for a combined license (COL) (e.g., a utility) may reference that NRC-reviewed design. Only their site-specific information and departures from the certified design need NRC evaluation. Once a design has been certified under 10 CFR 52, only a few, plant-specific design details would need NRC review. The efficiency of this approach is already reducing the time and effort required to issue each license.

The NRC has received 18 combined license applications to build and operate 28 new power reactors. As of the end of March 2012, four combined licenses have been issued and construction has begun at the respective sites.

TECHNICAL SESSIONS
Wednesday, March 14, 2012, 1:30 p.m. 3:00 p.m.

W8 *- Challenges and Lessons Learned in Design and Analysis of Civil Structures for New Reactors*

Session Chair: Mohammed Shuaibi, NRO

Session Coordinator: Milton Valentin, NRO
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The questions below were not answered during the above session:

Question 1: What is the expectation on the use of NDE (non-destructive examination) methods for steel-concrete (SC) composite components during construction? Will this be part of the ITAAC closure/review process?

Answer 1: The staff assumes that this question is related to the AP600 and AP1000 design, as these are the only designs certified by the NRC that use the composite steel-concrete construction approach. The AP1000 certified design described various construction and inspection approaches in the Design Certification Documents (DCD) and the Enhanced Shield Building Report that could be used for the inspection of the composite steel-concrete (SC) structural elements. These approaches included non-destructive evaluation (NDE) methods such as the Impact Echo Method, Ultrasonic Testing, and the Impulse Response Method. The choice of the particular technologies to use and the extent to which such technologies would be used in SC inspections will be determined by the Combined License holder, prior to construction, as part of the Combined License holder's commitment to develop a construction procedures program. These procedures will address inspection of modules pre-concrete and post-concrete placement. The NRC staff reviews the construction procedures program as part of the Combined License review. The Combined License holder will demonstrate the effectiveness of its inspection procedures of SC modules through the construction and inspection of mockups.

The inspections, tests, and analyses associated with the closure of building related ITAAC are included in the AP1000 DCD Tier 1 Table 3.3-6. Various construction inspection techniques are available and could be used to assess the adequacy of SC module construction. The extent to which NDE methods or a particular technology will be used in ITAAC related inspections will be informed by the construction and inspection of mockups. The NRC will use the Combined License holder's construction schedule to determine inspection opportunities. The results of the ITAAC related construction inspections will be used as the basis for the NRC's ITAAC Notice Closure Verification.

Question 2: What provisions (or alternatives) are available for (visual) inspection of SC structures after an earthquake as was conducted at North Anna on the exposed reinforced concrete after the earthquake which occurred there?

Answer 2: Regulatory Guide 1.166 "Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Actions" provides guidance for post-earthquake inspections. Various construction inspection techniques are available and could be used to assess the condition of SC structures post-earthquakes. The Combined License holder is required to develop inspection procedures as part of its construction procedures program and to test the adequacy of such inspection procedures and methods through mockup construction and tests. The extent to which a particular inspection approach, e.g. visual vs. NDE, will be used in a post-earthquake condition assessment will be informed by the results of the construction and inspection of mockups.

W9 *- Considerations for Long-Term Storage of Spent Nuclear Fuel on the Way to Disposal*

Session Chair: James Rubenstone, NMSS

Session Coordinator: Sandra Lindo-Talin, RES
301-251-7994, Sandra.Lindo-Talin@nrc.gov

The questions below were not answered during the above session:

Question 1 (addressed to Toshiari Saegusa, CRIEPI): Are we sure that sea water/salt water would be the worst case for stainless steel casks? Brackish water (salt, or non salt) at estuaries and ponds could be more corrosive. How can you be sure? Salt, pH, RH [relative humidity], and other impurities in brackish water need to be studied.

Answer 1: If the casks were contaminated by brackish water, the casks will be decontaminated. The cask surface will be kept free from the brackish water during storage in Japan. We should distinguish between canister and cask. The canister is an inner container in the concrete cask or horizontal concrete module. The stainless steel canister is a containment boundary that faces salty marine environment during storage. We should pay attention to potential loss of containment due to stress corrosion cracking by the marine environment.

Question 2 (addressed to Toshiari Saegusa, CRIEPI): In the leak test that measured for Kr [Krypton], doesn't such an indicator require cask seal degradation and fuel clad leakage?

Answer 2: Yes, presence of Kr will be indication of cask seal degradation and fuel clad leakage.

Question 3 (addressed to Toshiari Saegusa, CRIEPI): You mentioned the use of salt collection trays at the air inlet of concrete casks to mitigate chloride induced stress corrosion cracking. (a) Do these trays require periodic replacement or clearing to remain effective? (b) Is air flow significantly affected by salt filters? How often must salt filters be replaced?

Answer 3:

(a) These trays will require periodic replacement or clearing approximately once a year to remain effective.

(b) Temperature increase due to the salt collection tray system was evaluated; it was less than 0.5 degree in Centigrade (℃), so air flow does not appear to be significantly impacted.

Question 4 (addressed to Jeffrey Williams, DOE): How can small companies participate in the R&D -- DOE SBIR [Small Business Innovation Research], other? Small companies that are not current vendors can have out-of-the-box ideas.

Answer 4: The Department is working to create a synergy between the National Laboratories, Universities and private industry. There are two ways for private industry to participate in the future. All those seeking to participate should watch the FedBizOps.gov website for Requests for Proposals related to these activities. The second method is, if a company feels they have a specific idea or process that would greatly help the Department, an unsolicited proposal could be submitted. The National Energy Technology Laboratory in Morgantown, WV is the center for all unsolicited proposals for the Department. Information on how this could be done can be found at: http://www.netl.doe.gov/business/usp/unsol.html

Question 5 (addressed to Jeffrey Williams, DOE): What are the anticipated effects on the fuel cladding from the extended storage periods and then the vibration experienced during transport?

Answer 5: Studying the anticipated effects on the fuel cladding from the extended storage periods is one important subject of our ongoing research and development effort. The major factors being considered are oxidation, creep, annealing, and embrittlement, but the first three are expected to have less effect on the fuel cladding than embrittlement. The focus of our current study is on the nature, degree, timing, and consequences of possible embrittlement of the fuel cladding. Studying the anticipated effects on the fuel cladding from the subsequent vibration experienced during transport is another important subject of our ongoing research and development effort. The fuel assemblies are qualified for transportation when they're in new condition, and there is extensive experience with transporting them in recently used conditions. But there is little data on transportation after extended storage periods. DOE is studying the vibration loads on the fuel assemblies during transport and on all components of the fuel assemblies, including the fuel cladding.

Question 6 (addressed to Jeffrey Williams, DOE): What is DOE's timeline to bring its R&D activities to practice?

Answer 6: The Nuclear Energy Research and Development Roadmap indicates that R&D activities continue beyond 2020. The Roadmap covers R&D activities for four main objectives: 1) develop technologies and other solutions that can improve the reliability, sustain the safety, and extend the life of current reactors; 2) develop improvements in the affordability of new reactors to enable nuclear energy to help meet the Administration's energy security and climate change goals; 3) develop sustainable nuclear fuel cycles, and 4) understanding and minimization of risks of nuclear proliferation and terrorism.

With respect to near-term R&D, the Department is currently reviewing the January 2012 final report of the Blue Ribbon Commission (BRC) on America's Nuclear Future and will develop a strategy for the management of spent nuclear fuel and other nuclear waste within 6 months of publication of the BRC final report. The strategies cover both disposal and recycle fuel cycle options. All the activities needed to make progress for used fuel management are in the FY2013 budget request. The FY 2013 budget includes $60 million for R&D on storage, transportation, and disposal of nuclear waste to support the near-term recommendations put forward by the BRC. The budget request continues activities initiated in FY 2012 and specifically focuses on evaluating consolidated interim storage and transportation issues; working with industry to develop standardized approaches to used fuel management; conducting material testing to support extended storage of used fuel; revisit and prepare a report on plans to address recommendations identified by the National Academy of Sciences transportation report; and initiating research on geologic disposal alternative environments, e.g. system modeling, engineered barriers, natural barriers, evaluation of design concepts, and experiments.

Question 7 (addressed to Jeffrey Williams, DOE): Industry has successfully developed and deployed safe, secure, and efficient spent fuel dry storage technology. Regarding standard canisters, the time is not right until the end state requirements are defined. Given current budget deficits, and debt levels, shouldn't DOE just not spend the money?

Answer 7: The Blue Ribbon Commission on America's Nuclear Future, in its final report to the Secretary of Energy stated that *"…DOE should begin laying the groundwork for implementing consolidated storage and for improving the overall integration of storage as a planned part of the waste management system without further delay. Specific steps that DOE could take in the near term include:*

… Working with nuclear utilities, the nuclear industry, and other stakeholders to promote the better integration of storage into the waste management system, including standardization of dry cask storage systems. This effort should include development of the systems analyses needed to provide quantitative estimates of the system benefits of utility actions such as the use of standardized storage systems or agreements to deliver fuel outside the current OFF priority ranking. (These analyses would be needed to support the provision of incentives to utilities to undertake actions such as using standardized storage systems or renegotiating fuel acceptance contracts.)"

Consistent with this recommendation, the following was included in the FY 2012 DOE Office of Nuclear Energy budget directive: "Within available funds, $10,000,000 is for development and licensing of standardized transportation, aging, and disposition canisters and casks, but is not constrained to a single year". In implementing this direction, the DOE is currently developing a "Draft Implementation Plan on the Development and Licensing of Standardized Transportation, Aging, and Disposition Canisters". The approach taken will ensure a Systems Approach will be taken relative to the feasibility of Standardization. Inherent in this systems approach, DOE will document and consider assumptions with respect to the "end state requirements" and coordinate with both internal and external stakeholders before presenting resultant advantages and disadvantages to decision makers regarding the system benefits of standardization.

Question 8 (addressed to Jeffrey Williams, DOE): Are you seriously considering a storage canister that holds only four fuel assemblies? Isn't that inefficient and unreasonably burdensome on plant operators?

Answer 8: At present, no decision has been made on the viability of a canister containing a range of assemblies of used nuclear fuel. However, the DOE is currently studying the use of a wide range of possible sizes of canisters and one of the canisters being studied would hold 4/9 PWR/BWR assemblies. With respect to geologic disposal; concepts based on international programs were reviewed. These international disposal concepts are enclosed emplacement modes, whereby waste packages are in direct contact with encapsulating engineered or natural materials. Enclosed modes have less capacity to dissipate heat that open modes such as that proposed for a repository at Yucca Mountain. Thermal analysis has identified important relationships between waste package size and surface decay storage time needed to meet material temperature limits for these enclosed disposal concepts. Initial results indicate that a representative "four PWR [pressurized water reactor] fuel assembly disposal canister" will meet temperature limits for crystalline rock and clay/shale media and up to 12 fuel assemblies for salt media.

With respect to analyzing systems benefits of standardized canisters, our approach will ensure that the initial focus will be a Systems Approach to developing innovative solutions to address known disadvantages of a standardized canister system size which is any smaller than existing dry cask storage systems (24-37 PWR or 52-89 BWR [boiling water reactor] assemblies). Our plan will be to focus on first addressing the disadvantages of a small canister (with respect to cost, time, and dose issues at utility operating sites). If this initial work does not result in any innovation to solving the identified disadvantages of a small standardized canister size, DOE needs to make a decision regarding benefit/feasibility of continued pursuit of either small or large standardized canisters (given the known disposal issues with the large).

Question 9 (addressed to Jeffrey Williams, DOE): Is the DOE open to the possibility of considering the spent fuel canisters the retrievable waste form rather than the individual spent fuel assemblies?

Answer 9: The DOE is actively evaluating the possibility of considering the spent fuel canisters to be the retrievable waste form rather than the individual spent fuel assemblies. In most cases, because of the age of the fuel after extended storage, the effects of the fuel assemblies failing to perform their functions satisfactorily during normal transportation are not significant. So the canister could provide an adequate containment function and be considered the waste form under those conditions. But if there were a transportation accident where the canister was breached and water entered, the effects of the fuel assemblies failing to perform their functions satisfactorily during this hypothetical accident condition might be significant, since this could set up conditions for possible criticality – the self-sustaining level of radioactivity that powers nuclear generators. The DOE is performing two studies to evaluate solutions for this hypothetical accident condition.

The first study is evaluating what the risk of criticality would be if the fuel assemblies failed to perform their functions satisfactorily during a hypothetical accident and what programmatic changes or actions could be made to mitigate that risk to a regulatory acceptable level. The second study is to develop of a method to ensure that if there were a transportation accident either the canister would not breach and/or water could not enter, a concept called "moderator exclusion." Both of these studies are expected to provide preliminary results later this year.
Question 10 (addressed to Jeffrey Williams, DOE): What sort of numerical modeling simulations has DOE assembled for cask- canister – used fuel systems, and is there an expectation that these models can be used for NRC licensing activities?

Answer 10: DOE NE Fuel Cycle Research and Development's (FCRD) Used Fuel Disposition Campaign (UFDC) is generating a suite of limited capability generic performance assessment models and simulations that are environment specific for salt (layered/domal), granite / crystalline, clay / shale, and for deep borehole disposal system options. Generic design components have been devised for these individual evaluations; currently, no performance is allocated to the cask/canister/fuel in these generic simulations.

Modeling and simulation related to the engineered system components in question (cask/canister/used fuel) are being treated individually and collectively in rudimentary performance simulations to determine areas of importance to subsystem performance and those in need of improvement for future modeling efforts. More rigorous and advanced integrated modeling efforts are a mid-term and long-term goal of the NE FCRD/UFDC R&D effort (1-5, 5-10 years: coupled process models; thermal, hydrologic, chemical, mechanical) engineered (cask/canister) materials degradation, fuel/cladding degradation and characteristics (e.g., high burnup), and the role played by buffers and backfill are expected to be treated in these exercises.

The National Laboratories are directed to take necessary steps to ensure activities are conducted in accordance with applicable requirements as outlined in the Quality Assurance Program Document and as established in planning activities and in conduct of work such that the applicable information may be either directly used in future licensing (information may require qualification if it is to be used for licensing), or as support or corroborative information in the development of a future license.

Question 11 (addressed to Jeffrey Williams, DOE): Would you expect the fuel cladding testing program will answer the questions of fuel cladding integrity over extended period of storage, given the uncertainties of the environments the fuel is exposed to?

Answer 11: The testing and experimenting on cladding are not done with the objective of answering all possible questions about cladding integrity over the extended period of time of storage in certain environment. The idea of testing and experimenting on cladding is to understand the cladding behavior under environmental conditions which are most probable and/or bounding. This kind of testing and experimenting will provide us with information and data on hydride formation and its effects, annealing, creep, and corrosion resistance of the cladding. The data obtained will be used in the safety analysis for the used fuel and its container.

Question 12 (addressed to Jeffrey Williams, DOE): How long is it estimated to take to implement a standardized cask/canister/can-in-can concept? Will DOE license, build, design, and/or load these standardized systems at the reactor sites and at an MRS-like facility (for the stranded fuel)?

Answer 12: The Blue Ribbon Commission (BRC) for America's Nuclear Future, in its final report to the Secretary of Energy stated that *"…DOE should begin laying the groundwork for implementing consolidated storage and for improving the overall integration of storage as a planned part of the waste management system without further delay. Specific steps that DOE could take in the near term include:*

… Working with nuclear utilities, the nuclear industry, and other stakeholders to promote the better integration of storage into the waste management system, including standardization of dry cask storage systems. This effort should include development of the systems analyses needed to provide quantitative estimates of the system benefits of utility actions such as the use of standardized storage systems or agreements to deliver fuel outside the current OFF priority ranking. (These analyses would be needed to support the provision of incentives to utilities to undertake actions such as using standardized storage systems or renegotiating fuel acceptance contracts.)"

Consistent with this recommendation, the following was included in the FY 2012 Department of Energy (DOE) Office of Nuclear Energy budget directive: "Within available funds, $10,000,000 is for development and licensing of standardized transportation, aging, and disposition canisters and casks, but is not constrained to a single year". In implementing this direction, the DOE is currently developing a "Draft Implementation Plan on the Development and Licensing of Standardized Transportation, Aging, and Disposition Canisters". The approach taken will ensure a Systems Approach will be taken relative to the feasibility of Standardization. It is estimated that, within 2 years, the resultant advantages and disadvantages of standardization will be presented to decision makers regarding the system benefits of standardization. If the policy decision is to proceed with standardization, it is anticipated that SAR(s) will be submitted for the licensing of a Standardized Transportation, Aging, and Disposition Canisters within 4 years. As part of this Systems Approach, both delivery and loading at reactor sites and at an Independent Spent Fuel Storage Installation will be considered and analyzed.

Question 13 (addressed to Jeffrey Williams, DOE): How do you intend to examine prototypic high burnup fuel after it has been in storage for years? DOE has no hot cell that can open a prototypic dry cask. Plus NRC won't allow transportation of high burnup fuel in a dual-purpose canister from a utility site?

Answer 13: In order to examine prototypic high burnup fuel after it has been in storage for many years, studies will be done to address potential embrittlement of high-burnup fuel cladding by radial hydrides formed during drying/transfer operations and early stage of storage that affects post-storage transportation and fuel retrievability. There are plans to conduct tests and analyze, using existing equipment and available cladding specimens (both unirradiated/radial-hydride-treated and high-burnup Zry-4, ZIRLO, and M5), to help determine the processing parameters during drying/transfer operations, such as temperature, cooling rate, temperature cycles, etc., that would prevent formation of radial hydrides.

Also, there is an experimental foundation and knowledge base attained in the last decade on low-burnup fuel after 15 years of dry-cask storage, tensile and thermal creep studies of high-burnup (67 GWd/MTU) cladding, and more recently, radial-hydride formation and radial-hydride-induced embrittlement in cladding as a function of drying-transfer conditions. It is anticipated that facilities will be available in the future for handling stored used nuclear fuel when needed.

Question 14: The industry has had to use the courts to receive payment for dry fuel storage. When will, or why won't the government create a process to cover these costs using the spent fuel funds, paid for by the rate payers, without requirement for litigation?

Answer 14: The issue in this question lies outside of the NRC's regulatory authority. This issue was addressed by the Blue Ribbon Commission (BRC) on America's Nuclear Future in their final recommendations to the Secretary of Energy (www.brc.gov). One of the recommendations of the BRC was to ensure access to the funds nuclear utility ratepayers are providing for the purpose of nuclear waste management.

Answer 14 (addressed to Jeffrey Williams, DOE): The Department of Energy is currently evaluating the recommendations from the Blue Ribbon Commission on America's Nuclear Future report issued in January 2012. The report has addressed the nuclear waste management issues facing the nation. Pending assessment and resolution of a path forward many of the issues will be clarified in the future. Part of this path forward will include how the program should be funded.

Question 15: With respect to retrievability, are there any differences expected between BWR or PWR UO_2 fuel and MOX fuel?

Answer 15: Assessments of the retrievability of stored spent fuel, whether UO_2 or MOX, are conducted as part of the certification and renewal of spent fuel storage systems, and any fuel-specific considerations are addressed in these assessments. Several factors are considered in evaluation of fuel behavior during storage. Potential differences between UO_2 fuel and MOX fuel include differences in cladding materials, operating history, and radionuclide composition of the spent fuel. Such differences could lead to, for example, differences in the decay heat generation rate. Because many degradation mechanisms are temperature dependent, differences in fuel temperature over time, coupled with differences in materials and operating histories, could lead to differences in degradation of fuel or hardware that could affect retrievabilty. NRC is currently examining how long-term degradation mechanisms could affect future performance in dry storage, including regulatory requirements for retrievability, depending on the specific nature of the fuel.

Answer 15 (addressed to Jeffrey Williams, DOE): There would be no differences between BWR and PWR UO_2 fuel and MOX fuel retrievability. However there are differences for both BWR and PWR if they were high-burnup UFD, as follows. In dry storage, the dry cask has to provide safe confinement/containment and, in parallel, the decay heat has to be removed to limit temperature induced material alterations. This means, dry storage is more sensitive to UOX & MOX high- burnup in wet storage where higher temperatures and, consequently, higher stresses on the cladding. The ability to meet applicable regulatory limits will need to be re-evaluated for higher burnup UOX and MOX. The result of these evaluations may require, for example, a redesign of the cask heat removal and shielding systems, a decrease in the number of spent fuel assemblies than can be placed into a single storage cask, and an increased decay time in the pool prior to placement in dry storage.

Question 16: I realize there has long been political opposition for it, but what are the scientific risks or consequences to reprocessing spent fuel?

Answer 16: Reprocessing of spent nuclear fuel would involve the management of both chemical and radiological hazards. The actual risks, in the sense of both the hazards and the frequency of human or environmental exposure to those hazards, would depend upon the specific design of the reprocessing plants, including its safety systems, and the manner in which it was operated. Should such a facility be licensed in the United States, NRC regulations and oversight would be in place to ensure public health and safety, security, and protection of the environment.

Answer 16 (addressed to Jeffrey Williams, DOE): The technology needed to reprocess exists today. Even though the technology exists, reprocessing requires very specialized equipment and facilities and is a complex and involved activity that can produce mixed hazardous waste streams. As with any industrial process, nuclear, industrial and safety risks exist during the reprocessing activities.

Reprocessing is an option that may or may not fit into the near term overall used fuel management policies. The Department is evaluating this and other options that were presented by the BRC. The Department is still in the process of evaluating the BRC recommendations to determine the best path forward. When this is completed, the Department will work with Congress and other stakeholders to develop a workable solution for the future.

Question 17: Since Yucca Mountain is no longer an option, is there another national alternative for permanent spent fuel storage?

Answer 17: On January 29, 2010, President Obama directed the Secretary of Energy to establish the Blue Ribbon Commission (BRC) on America's Nuclear Future to conduct a comprehensive review of policies for managing the back end of the nuclear fuel cycle and recommend a new strategy. Pursuant to its Charter, the BRC provided its final recommendations to the Secretary of Energy on January 26, 2012 (www.brc.gov). The eight key elements of the BRC recommended strategy are:

1. A new, consent-based approach to siting future nuclear waste management facilities.

2. A new organization dedicated solely to implementing the waste management program and empowered with the authority and resources to succeed.

3. Access to the funds nuclear utility ratepayers are providing for the purpose of nuclear waste management.

4. Prompt efforts to develop one or more geologic disposal facilities.

5. Prompt efforts to develop one or more consolidated storage facilities.

6. Prompt efforts to prepare for the eventual large-scale transport of spent nuclear fuel and high-level waste to consolidated storage and disposal facilities when such facilities become available.

7. Support for continued U.S. innovation in nuclear energy technology and for workforce development.

8. Active U.S. leadership in international efforts to address safety, waste management, non-proliferation, and security concerns.

Several of the BRC recommendations are related to ongoing areas of NRC regulatory activities and NRC is positioned to support national policy changes in areas associated with its regulatory purview.

Answer 17 (addressed to Jeffrey Williams, DOE): In January 2010, Secretary of Energy Chu formed the Blue Ribbon Commission (BRC) on America's Nuclear Future to provide recommendations for developing a safe, long-term solution to managing our Nation's used nuclear fuel and nuclear waste. BRC conducted a comprehensive review of policies for managing the back end of the nuclear fuel cycle, in the United States, and abroad, and issued their final report in January 2012.

The Department is currently reviewing the January 2012 final report of the BRC on America's Nuclear Future and, as directed by the "Consolidated Appropriations Act, 2012, will develop a strategy for the management of spent nuclear fuel and other nuclear waste within 6 months of publication of the BRC final report. The strategies cover both disposal and recycle fuel cycle options. At this point, there is not a national alternative for permanent spent fuel storage.

Question 18: If chloride cracking of stainless steel is such a problem, why not change canister material or use double-wall rather than living with it?

Answer 18: NRC evaluates the designs of dry storage systems that are proposed by applicant; these designs use stainless steel. NRC's role is to ensure that such approved designs continue to perform their regulatory functions in the relevant environments. Stress corrosion cracking (i.e., chloride cracking) of stainless steel canisters has not yet been shown to be a major problem in dry cask storage. As the speakers discussed during the session, uncertainties exist as to the conditions where stress corrosion cracking can occur, and if the canisters used in dry storage experience those conditions. The NRC and industry have initiated research and inspection tasks to address aspects of this issue.

Answer 18 (addressed to Jeffrey Williams, DOE): The environments that canister materials are exposed to during storage are far less aggressive than what testing programs consider. Stainless steels in general are good corrosion resistance materials for canisters. The chloride concentrations for assumed deliquescence conditions affect only a small portion of the canister surface and the overall structural integrity of the canister is not expected to be compromised. The test data being generated will inform the design and a robust design will be implemented.

Question 19 (addressed to John Kessler, EPRI): With engineering jobs growing at a greater rate than students pursuing engineering degrees, what is the plan for the far-future of spent fuel, i.e. 120 years from now.

Answer 19: For 120 years from now, of course, there is not plan at the moment. For the next few decades, however, the nuclear industry is funding a wide variety of education programs to ensure the continuation of a trained nuclear work force. Industry provides grants to academic institutions ranging from establishing nuclear curricula at community colleges to grants and scholarships at the undergraduate university level. Equally important are industry's efforts to develop training programs for skilled trades in the nuclear industry, such as nuclear-qualified welders. Industry is also engaging regulators to extend the licenses of existing plants out to 80 years. This will require the maintenance of a skilled workforce for existing reactor designs and operations.

In the longer run, the need for industry support for future students will depend on the type of nuclear reactors in use (e.g., continued primary use of light water reactors with a transition to Generation-III reactor designs; transition to a combination of fast and light water reactors, or some other combination. For such a future, we will need to maintain the skill set to continue to safely operate LWRs, plus a continued, major effort to develop commercial-scale fast reactors. For fast reactors, some research and a lot of "development" in terms of engineering needs to be done. This will require expanding nuclear engineering and related engineering programs to address the complex engineering problems related to reliable, commercial-scale fast reactors. The eventual introduction of reprocessing will also involve nuclear and chemical engineers and a large number of trained crafts people and operators.

Question 20 (addressed to John Kessler, EPRI): What is the current industry perspective on the use of spent fuel pools beyond 60 years?

Answer 20: As briefly discussed in the response to the previous question, there is an ongoing effort to develop the technical bases for continued operation of nuclear plants out to a total of 80 years of operation. The spent fuel pool is an integral part of the reactors, so demonstrating continued, safe performance of the spent fuel pool also is being considered as part of the license extension activities. In particular, a plant-wide aging management plan is being developed that includes a combination of inspection, mitigation, and repair, as appropriate. Currently, EPRI is engaged in developing NDE techniques to identify cracks in the welds joining the stainless steel plates that form the spent fuel pool liner. EPRI is also exploring the use of weld overlays to repair those sections of the pool liner weldments. EPRI is also involved in activities to evaluate the condition of the concrete behind the spent fuel pool liner that provides the structural support.

Question 21 (addressed to John Kessler, EPRI): How about radiation assisted corrosion and the compound effect of irradiation-assisted corrosion and SCC in marine environment?

Answer 21: Yes, all forms of corrosion need to at least be considered. There is a significant amount of literature on irradiaton-assisted stress corrosion cracking (IASCC) of stainless steels upon which to draw. These effects need to be considered regarding their potential importance of SS corrosion in marine environments.

Question 22 (addressed to John Kessler, EPRI): In the past, there was discussion of performing "accelerated aging" tests, what exactly was that and why is it not longer discussed as an option?

Answer 22: Accelerated aging testing is very much alive and well. Accelerated aging testing is meant to evaluate the degradation mechanisms and potential consequences of long-term storage system degradation over reasonable laboratory time scales (rather than having to run an experiment for years or even decades). For example, some of the CRIEPI work presented by Dr. Saegusa was under "accelerated" conditions (use of higher temperatures in combination with highly concentrated salt brines). For example, for temperature-dependent corrosion, higher (than under actual conditions) temperature testing is done at a few temperatures to ascertain the activation energy involved in the corrosion reaction. That activation energy can then be extrapolated to lower temperatures using an Arrhenius function.

Accelerated corrosion testing needs to be performed carefully, however, as the accelerated conditions may introduce and/or obscure degradation mechanisms that would not occur under actual, long-term conditions.

Question 23 (addressed to John Kessler, EPRI): Are there efforts in place to determine certifiability for transport of storage-transport casks? This could involve NRC, DOE, and industry coordination.

Answer 23: Yes, efforts have been underway for many decades on this subject. Indeed, the R&D to collect the data to establish the technical bases for safe transportation of spent fuel has been performed by industry, DOE, and NRC. This work includes, for example, understanding the mechanical behavior of spent fuel cladding to evaluate the amount of damage the spent fuel might incur in a transportation accident. Many studies involving maintaining subcriticality of a flooded transportation cask (an NRC-required assumption to obtain a transportation certificate) have been performed that has led to the use of neutron absorbers inside the transportation casks and the limited use of burnup credit.

Industry and NRC have had cooperative agreements to share in spent fuel transportation data collection for many years. There are additional DOE-NRC and DOE-industry co-funded R&D activities in this area, as well.

Question 24: re: NRC 12/30/11 release of Waste Confidence Update. How can NRC rely on the BRC's report in advance of DOE's responses this summer, and congressional approval of any actions DOE would implement – isn't this presumptuous of the NRC, since no appropriations for such actions have been approved?

Answer 24: The NRC is not relying on the BRC report, though it may reference the BRC recommendations and subsequent developments. The NRC has already stated that the EIS would need to account for new, relevant laws or policies that may be established in the coming years. The NRC staff's planned approach for the EIS, as discussed in the 12/30/11 document on background and preliminary assumptions, includes both geologic disposal and consolidated storage as part of future waste management scenarios. The EIS for the Waste Confidence update will assume geologic disposal is the endpoint for spent nuclear fuel, because that remains the requirement under the Nuclear Waste Policy Act. Consolidated storage is included in the proposed EIS scope as a reasonable alternative to storing spent fuel at reactor sites for extended periods. Both of these aspects are consistent with the BRC report but are not tied to specific implementation of their recommendations.

Question 25: DOE and EPRI speakers discussed the need for much more research to justify the long-term storage and subsequent transportation of SNF. In light of those needs, how does NRC justify the confidence decision/rule?

Answer 25: The basis for the current Waste Confidence decision and rule is contained in the statements of consideration for the 2010 update. See 75 *Federal Register* 82032 and 81037 (December 23, 2010).

Question 26: How much more time does industry have before the first UNF [used nuclear fuel] reaches the 120 year limit of the existing Waste Confidence Rule? Will planned testing, and its planned time-frame ensure this UNF does not go "out of scope" with respect to the 120 yrs? If not, then what, exemptions?

Answer 26: The Waste Confidence Rule is a generic determination of no significant environmental impacts from spent fuel storage for the stated time period. It is not a regulation for spent fuel storage facilities. Dry storage of spent fuel is subject to the safety and security requirements in 10 CFR Part 72. Dresden 1, licensed in 1959, is the earliest licensed power reactor; 60 years beyond licensed life for Dresden 1 is 2059. The Dresden license for dry storage under 10 CFR Part 72 was issued in 2000. NRC's regulatory program for dry storage, including aging management, is intended to be protective of public health and safety for fuel stored for extended periods. The staff continues its work to understand the technical and regulatory issues associated with extended storage and will be proposing regulatory changes if necessary.

Question 27: Should or will moving UNF to either a repository or a consolidated interim storage facility be regulated by NRC or DOE? Same questions as above, except for HLW [high level waste] and GTCC [Greater than Class C waste] and naval fuels.

Answer 27: Federal regulation of spent fuel transportation safety is shared by the U.S. Department of Transportation and the NRC. The NRC carries out its regulatory responsibilities through a combination of safety and security requirements, certification of transportation casks, inspections, and a system of monitoring to ensure that requirements are being met. The Nuclear Waste Policy Act of 1982 as amended, required DOE to use casks certified by NRC when it transports spent fuel to a national high-level waste repository or monitored retrievable storage facility.

W10 *- Flooding: Lessons Learned and Near-Term Regulatory Actions*

Session Chair: Scott Flanders, NRO

Session Coordinator: Barry Miller, NRR
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For questions and answers from the RIC as well as additional information on this topic, please contact the Session Coordinator listed above.

W11 *- Decommissioning Planning including Funding Assurance: A Discussion on Policies and Practices for Implementing the Decommissioning Planning and License Termination Rules; Regulatory Guides 4.21 and 4.22*

Session Chairs: Thomas Fredrichs, NRR, James Shepherd, FSME

Session Coordinators: Kosmas Lois, NRR
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Shawn Harwell, NRR

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The questions below were not answered during the above session:

Question 1: What is the due date for the first submittal of the updated decommissioning funding plans required under amendments of the Decommissioning Planning Rule issued on June 17, 2011? (76 FR 35512)

Answer 1: The due date for the first submittal of the updated decommissioning funding plans is December 17, 2012, the effective date of the Decommissioning Planning Rule, as discussed in the Responses to Comments H.24 and H.25.3 in the Notice of Final Rulemaking (76 FR 35512, 35550 - 51). The due date applies to licenses issued under Parts 30, 40, 70, and 72. Holders of licenses issued under those Parts should review their decommissioning funding plans and submit revisions, if necessary to comply with the amended requirements, by December 17, 2012. The requirement applies to both specific and general ISFSI licenses issued under 10 CFR Part 72. For holders of power reactor licensees issued under 10 CFR Part 50 that have submitted a site-specific cost estimate following the permanent cessation of operations, as required under 10 CFR 50.82(a)(4)(i), the due date for their first submittal of the status reports for financial assurance and funding for irradiated fuel management, based on the new requirements in 10 CFR 50.82(a)(8), is March 31, 2013. As stated in Section M of the Supplementary Information to the Final Rule (76 FR 35512, 35521), permanently shutdown reactor licensees must continue to submit a decommissioning funding status report as required under 10 CFR 50.75(f)(1) and (2).

W12 *- Implementing the IAEA Action Plan: Working Together Flooding: Lessons Learned and Near-Term Regulatory Actions*

Session Chair: Margaret Doane, OIP

Session Coordinators: Danielle Emche, OIP
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For questions and answers from the RIC as well as additional information on this topic, please contact the Session Coordinators listed above.

W13 *- Proposed Emergency Preparedness Regulatory Activities and Incident Response Improvements Following Real-World Events*

Session Chair: Mark Thaggard, NSIR

Session Coordinators: Bethany Cecere, NSIR
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Barry Miller, NRO
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The questions below were not answered during the above session:

Question 1: For NTTF 9.3, you stated that licensees need to tell you about interim actions to be taken following completion of the staffing assessments for multi-unit events or to implement 4.2. What do you envision for interim actions by licensees? Timing?

Answer 1: Response provided by NRC staff: First, as a point of clarification, the interim actions taken would have been determined by a licensee prior to conducting the staffing assessment. In discussions with Industry, it was recognized that licensees may have initiated steps to frame any lessons learned from the accident at Fukushima and thereby determined some strategies to implement prior to making any procedural changes. The NRC has not defined any expected interim actions however, the NRC is interested in whether any actions have been taken. The NRC expects that these activities would be included in the 90-day response to the 50.54(f) letters.

Question 2: Any thoughts on Social Media and impact on ability to provide factual data to state/NRC/local officials/all stakeholders in a timely manner? Social Media, CNN, etc. get info out quicker than facility capabilities.

Answer 2: Response provided by Ms. Perkins-Grew (NEI): Use of social media platforms as a means to provide data to all stakeholders in a timely manner is extremely important and is an absolute necessity in the current information sharing environment. NEI is leading an industry task force to ensure that we, as an industry, are better positioned to leverage the social media as a primary means for communicating event information as efficiently as possible. This will include the use of social media as an additional communications tool to augment industry's currently emergency response communications capabilities. NEI is also leading a second task force to develop guidelines for industry to establish a "Joint Information System" to improve upon traditional means of communicating emergency information that is focused on a single facility versus a system of capabilities.

Question 3: You mentioned in your presentation that industry was not clear on government roles in emergency response. Is this more a case of lack of specific info to find or not educating oneself with existing info out there? (Communication challenge with government?)

Answer 3: Response provided by Ms. Perkins-Grew (NEI): This was attributed to a lack of familiarity with roles of agencies other than the NRC in addition to the interrelationships within the National Response Framework. This was further complicated by the fact that the event occurred outside of the U.S.

Question 4: The lessons learned that you discussed are based on the event in Japan. Had this event occurred in the U.S., do you believe the demands on the U.S. industry would have been greater? Are you addressing the recommendation in a broad enough manner to meet the increased demands of a domestic event?

Answer 4: Response provided by Ms. Perkins-Grew (NEI): Yes, we are certain that the demands on the U.S. industry would be greater. That is why NEI, INPO, and EPRI are actively engaged in improving our response capabilities to support industry. Based on the lessons learned from their response to Fukushima, the three organizations have developed a new “framework” document further clarifying responses to both domestic and international events.

Question 5: Lack of free flow of information during Fukushima caused lots of confusion. How can this issue can be helped with increase information security you are suggesting?

Answer 5: Response provided by Ms. Perkins-Grew (NEI): At the onset of the event, yes, there was a lack of free flow of information. However, as the event progressed information was being shared in some cases without sensitivity to whether the information was company-sensitive. The information security that I was referring to was in the context of sharing information among the industry responders to vet and validate prior to release to stakeholders. Action items based on lessons learned from the response to Fukushima include a review of processes and protocols for developing protected Web sites for sharing proprietary, sensitive information. The objective is to ensure the sharing of appropriate information during an event is unimpeded.

Question 6: It is not clear what is within scope of 50.54(f) letter regarding multi-unit staffing for beyond DBA. Currently, we are only staffed on shift to address a single unit event (DBA) except for LOOP (coincident with single unit DBA). Are we now expected to be staffed for multiple DBA and beyond DBA?

Answer 6: Response provided by NRC Staff: The purpose of the 50.54(f) letters is to solicit information from licensees to determine whether to modify existing licenses to ensure that licensees can effectively respond to and mitigate the following event; a large scale natural event that causes the sites (all units) to lose all AC power with impediments to the units. The expectation from the 50.54(f) request is to obtain the information the agency needs to determine whether licensees’ onsite staff, coupled with the augmented staff, is capable of handling such an event or what staff should be added onsite. In addition, the NRC will be requiring new coping strategies and analysis to determine if additional staff is warranted as outlined by the rulemaking efforts for station blackout, severe accident management guidelines (SAMGs), and the orders for Near-Term Task Force Recommendation 4.2 (mitigation strategies).

Question 7: Do you have a preliminary estimate of number of regional warehouses needed to respond timely to an emergency? Thank you.

Answer 7: Response provided by Ms. Perkins-Grew (NEI): Once the industry survey is completed, INPO will have a better idea. More than one and less than five would be an estimate.

Question 8: During Fukushima, the NRC Operations Center received many calls requesting immediately important information e.g. can airlines still fly to Japan. The NRC forwarded these calls to EPA, but no real information was given. Has this communications bottleneck been repaired and who is this contact?

Answer 8: Response provided by NRC Staff: The NRC conducted a post-event response After Action Review that, among other things, addressed needed enhancements to our real-time communications with stakeholders during events. The After Action Review items are currently being prioritized and scheduled for both short term and long term resolution by our Incident Response staff.

Question 9: Given an event of regional influence in a domestic setting, what agency will have the lead role in the transport aspect of the staged equipment? (Taking into account potential damage to transportation infrastructure.)

Answer 9: Response provided by Mr. Sherwood (FEMA): Primary responsibility for management of incidents involving transportation normally rests with State and local authorities and the private sector, which own and operate the majority of the Nation's transportation resources. As such, a Federal response must acknowledge State and local transportation policies, authorities, and plans that manage transportation systems and prioritize the movement of relief personnel and supplies during emergencies.

Under the National Response Framework, the Secretary of Transportation coordinates ESF #1 - Transportation, consistent with DOT's statutory mission, to promote fast, safe, efficient, and convenient transportation in support of the national objectives of general welfare, economic growth and stability, and the security of the United States.

DHS/Federal Emergency Management Agency (FEMA) is responsible for the provision of transportation assets and services (including contracts or other agreements for transportation assistance) for responders, equipment, and goods, consistent with the ESF #7 - Logistics Management and Resource Support Annex.

The Director of Military Support (DOMS) is the responsible national-level Department of Defense (DOD) office for military support to civilian authorities. DOD has responsibility for ESF #3 - Public Works and Engineering, and has designated the U.S. Army Corps of Engineers (USACE) as its operating agent for ESF #3 planning, preparedness, response, and recovery. Activities within the scope of ESF #3 include: Emergency clearance of debris to enable reconnaissance of the damaged areas and passage of emergency personnel and equipment for lifesaving, property protection, and health and safety, and provision of expedient emergency access routes, which includes repairs to damaged streets, bridges, ports, waterways, airfields, and other facilities necessary for emergency access to disaster victims.

W14 *- The Applicability and Use of Third-Party Information for Operating Reactors*

Session Chair: Michael Markley, RR Margaret Doane, OIP

Session Coordinator: Andrew Hon, NRR
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Questions submitted during the above session were answered during the session's Q/A period.

TECHNICAL SESSIONS
Wednesday, March 14, 2012, 3:30 p.m. 5:00 p.m.

W15 *- Foreign Ownership, Control, or Domination*

Session Chair: Christopher Regan, NRR

Session Coordinators: Jo Ann Simpson, NRR
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Anneliese Simmons, NRR

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The questions below were not answered during the above session:

Question 1: Since there is no numerical threshold, what is the standard for acceptance?

Answer 1: As stated in the "Final Standard Review Plan on Foreign Ownership, Control or Domination," (64 FR 52355, September 28, 1999), (the SRP), the Commission has not established a specific threshold above which it would be conclusive that an applicant is controlled by foreign interests. The SRP further states that "an applicant is considered to be foreign owned, controlled or dominated whenever a foreign interest has the "power," direct or indirect, whether or not exercised, to direct or decide matters affecting the management or operations of the applicant". In response, the NRC reviews potential foreign ownership, control, or domination (FOCD) issues on a case-by-case basis based on the facts and circumstances of the application and supplemental information.

Question 2: Would the NRC consider any unique treatment to the scenario where a Part 52 COL is issued but not acted on for an extended period? Could there be exemptions/license conditions in place for this period? That is, FOCD would need to be resolved before construction begins.

Answer 2: As required by Section 103d of the Atomic Energy Act of 1954, as amended, the FOCD determination must be made at the time the combined license (COL) is issued. Because this is a statutory requirement, the NRC cannot grant an exemption from it. The FOCD determination is based on the particular facts and circumstances at the time of application, and the use of license conditions or other negation measures would have to be evaluated based on the specific circumstances presented. If the licensee's FOCD status changes at any time, the NRC staff will perform a new FOCD review either independently or as part of a license transfer review.

Question 3: Do you see any clear difference between the FOCD review for reactor licenses and for materials licenses, such as uranium mining or refinement?

Answer 3: The prohibition of FOCD for Part 50 licensees is a requirement of Section 103 of the Atomic Energy Act of 1954, as amended. The FOCD review based on the SRP of 1999 pertains only to Part 50 licensees and Part 52 COL applicants. Materials licensees are not subject to a similar FOCD review, but are still subject to a finding that issuance or transfer of the license is consistent with the common defense and security.

Question 4: Should a publicly traded licensee request a FOCD review, if it learns that a foreign entity has: 1) acquired a small equity interest (e.g., 1%); or 2) has purchased a significant amount of its debt securities on the open market? No other involvement from the purchaser – licensee has no active licensing actions before the NRC.

Answer 4: Again, there is no safe harbor for FOCD and whether a publicly traded licensee should request an FOCD review would depend on the facts and circumstances of the situation. The licensee should refer to "Final Standard Review Plan on Foreign Ownership, Control or Domination," (64 FR 52355, September 28, 1999) for guidance on FOCD issues.

Question 5: What other agencies perform foreign ownership reviews?

Answer 5: A number of Federal Agencies perform foreign ownership reviews. For example, the Federal Communications Commission regulates the telecommunications industry; the Department of Transportation limits foreign ownership for airlines; foreign investors in critical infrastructure are reviewed by the Committee on Foreign Investment in the United States (CFIUS); and the Department of Defense, Department of Energy, and others perform foreign ownership reviews under the National Industrial Security Program Operating Manual (NISPOM) on a case-by-case basis.

W16 *- Near-Term Issues Related to Spent Fuel Pools*

Session Chair: Richard Lee, RES

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

W17 *– Regional Administrators' Session on Current Regional Issues*

Session Chair: Martin Virgilio, OEDO

Session Coordinator: Gerald McCoy, RII
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The questions below were not answered during the above session:

Question 1: Since security findings/violation SDP process is deterministic, how can we say it's been brought back into the ROP? I would vote to not reflect colors in security because of this.

Answer 1: The security cornerstone has always remained part of the ROP framework, even though a separate assessment program has been applied to security inspection findings and performance indicators since 2005. The security cornerstone is not unique in its application of deterministic significance determination tools. The two radiation safety cornerstones and the emergency preparedness cornerstone similarly apply deterministic significance determination tools that distinguish among green, white, yellow and red significance levels. The staff has not proposed, and the Commission has not approved, a change in policy governing the disclosure of sensitive unclassified non-safeguards information (including actual significance levels or colors associated with security inspection findings and performance indicators). As such, colors used in this cornerstone will be either green (to represent performance of very low safety significance) or blue (to represent inspection findings and performance indicators that are of low to high safety significance).

Question 2: Does the reintegration of security issues into the ROP signal any change in the agency's view of what constitutes security sensitive information under 10 CFR 2.390?

Answer 2: No, it does not.

Question 3: Discuss NRC's position or understanding of how industry is dealing with the January 2012 Byron switchyard event, including reporting.

Answer 3: The staff issued an Information Notice on the potential design vulnerability on March 1, 2012. The staff is continuing to review the event and the design basis for Byron, as well as monitoring industry actions, but has not come to any regulatory decisions.

Question 5: Inconsistency in citing cross-cutting findings for SLIV findings. Some inspectors say it was evaluated under traditional enforcement so cross-cutting does not apply. Other residents cite a cross-cutting finding. Doesn't seem to be confined to one specific region, but to specific inspectors. This makes it difficult to develop meaningful metrics in cross-cutting areas. It is difficult to respond to this question without more specific information.

Answer 5: The guidance in IMC 0612 is intended to ensure reliable outcomes, and cross-cutting issues are a focus of regional reliability initiatives. Cross-cutting aspects are assigned only to inspection findings in accordance with the guidance in IMC 0612 Appendix B. Cross-cutting aspects are not assigned in accordance with the Enforcement Policy which is used to disposition traditional enforcement violations (i.e., those violations with a severity level).

However, some SL-IV violations may be associated with an ROP finding stemming from the same more than minor performance deficiency and as such may be assigned a cross cutting aspect since both the Enforcement Policy and IMC 0612 Appendix B apply.

Question 6: Cross-cutting aspects are required to be different than the cited finding or violation. This creates a challenge since the causal analysis conducted by the licensee typically aligns to the basic criteria for the performance deficiency, thus we have causal analyses that are not providing insights for the cross cutting aspect. Why is this?

Answer 6: Cross-cutting aspects are *not* required to be different from the cited finding (not violation, since they are not assigned to violations). A cross-cutting aspect is defined in IMC 0310 as a performance characteristic of a finding that is the most significant causal factor of the performance deficiency.

Question 7: Introduction of substantive cross-cutting issues into the ROP has contaminated the overall objective of the ROP. Currently ROP has become extremely subjective as a result of rough contamination. What is the NRC doing to improve ROP to make it more objective rather than subjective in the hands of resident inspectors and/or opinion of the respective region? It might be a solution to introduce uniformity of enforcement of regulation among various regions.

Answer 7: The substantive cross-cutting issue (SCCI) process is being implemented consistent with the Commission's direction to more fully address safety culture through the use of cross-cutting issues, which have been represented in the ROP since its inception in 2000. The ROP was designed to be an objective, risk-informed performance assessment process. By contrast, determinations regarding safety culture *by their intrinsic nature* warrant some degree of subjective judgment. The staff is aware of discontent with the SCCI process among some internal and external stakeholders.

The ROP framework provides the objective process for assigning cross-cutting aspects and developing SCCIs. The same process applies to all regional offices and all licensees. Since the SCCI process provides indications of performance in cross-cutting areas within the structured ROP framework, those indications are developed and characterized as performance deficiencies in accordance with the fundamental regulatory principles of the ROP. Findings with safety culture aspects are developed and characterized in a manner that is as objective as is practicable within the ROP; they also are transparent, understandable, predictable, risk-informed and performance-based. Furthermore, to be documented in an NRC inspection report and be assigned a cross-cutting aspect, the underlying performance deficiencies are of more than minor significance and their significance is risk-informed.

W18 *- Regulatory Actions on Extended Station Blackout Events*

Session Chair: Michael Cheok, NRR

Session Coordinator: Barry Miller, NRR
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Questions submitted during the above session were answered during the session's Q/A period.

W19 *- Regulatory and Safety Applications of International Safety-Significant Operating Experience*

Session Chair: Michael Cullingford, NRR

Session Coordinator: Susan Wittick, OCA
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Questions submitted during the above session were answered during the session's Q/A period.

W20 *- Small Modular Reactors: A Discussion of Safety, Security, Environmental, and Economic Issues from a Variety of Viewpoints*

Session Chair: Stephanie Coffin, NRO

Session Coordinator: Jan Mazza, NRO
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The questions below were not answered during the above session:

Question 1 (addressed to TVA/mPower): Do you take into account the district heating application of SMRs abroad? It depends on distance from the city = exclusive zone diameter = safety parameters

Answer 1: District heating is being considered as a potentially important application, particularly for certain overseas users. Siting criteria, including exclusion areas and low population zones, are important considerations in such an application. GmP is working with U.S. and international customers and regulators to ensure such siting criteria are appropriately risk-informed to take advantage of mPower's intrinsic design attributes.

Question 2 (addressed to TVA/mPower): What are the plans for removing used fuel from plant?

Answer 2: The mPower design uses standard, reliable fuel movement methods and equipment; sizing of used fuel storage is based on industry needs and customer feedback, and the mPower design includes ample provisions for transfer to long-term safe, dry storage.

Question 3 (addressed to TVA/mPower): Could you elaborate a little on the “reactor control” test aspect of your IST at Lynchburg?

Answer: The IST contains systems of sufficient fidelity such as core (electric heaters), once thru steam generator and main steam/main feedwater and a programmable control system (including heater power control) to permit testing and refinement of control strategies and protection envelope appropriate to the mPower plant. The heater power supply has a point kinetics capability to adjust the power including loop water temperature effects so as to simulate the changes in "reactivity”.

Question 4 (addressed to TVA/mPower): Since the SMR is underground, how will you mitigate the risk caused by a tsunami or a flood for an extended period?

Answer 4: The most straight forward approach to protecting a plant from an external flood is setting the plant grade elevation above the maximum flood elevation resulting from the worse case event for a given site. The worse case evaluation should consider the effects of local precipitation, flooding of streams and rivers, dam breaches, storm surges, seiches, tsunamis, ice induced flooding, and channel migration or diversion. In addition, the design incorporates reliable hardened protective features, such as dikes, levees, flood doors, submarine hatches, waterstops at construction joints, and pipe penetration seals.

Question 5 (addressed to TVA/mPower): Which design feature of mPower is challenging to get licensed?

Answer 5: The GmP team is working with the NEI industry team to target key areas of regulatory infrastructure in the interest of risk-informing regulations and guidance to take advantage of smaller, passive designs. Risk-informed decisions on staffing, emergency planning, etc., are important strategic focus areas. There are aspects of the mPower design that are innovative, but they are evolutionary and based on existing reliable concepts and components, or the subject of rigorous testing and significant pre-application interactions with the NRC staff. While we anticipate a robust, comprehensive review by the NRC staff, we have not identified significant discrete challenges to licensing the mPower design.

Question 6 (addressed to TVA/mPower): What is the feasibility and possibility of the DOD being the first customer to build and operate an SMR?

Answer 6: GmP is working with a number of prospective customers, and believes that the opportunity to provide dedicated power for national labs, military installations, etc., is viable and could be important in supporting strategic national assets and infrastructure. A number of DOD and DOE facilities have expressed interest in possible SMR deployments. It is very possible that one of these installations will be first to benefit from the clean and reliable electric supply provided by SMR technology.

Question 7 (addressed to TVA/mPower): What is Generation mPower's role in the current design efforts by B&W?

Answer 7: Generation mPower LLC (GmP) is a joint company formed by the Babcock & Wilcox Company (B&W) and Bechtel Power Corp. to design, license and build the next generation of nuclear power plants based on B&W mPowerTM reactor technology. GmP directs overall efforts of B&W and Bechtel in developing and licensing the mPower design, and will ultimately be the single point plant supplier to utility customers.

Question 8 (addressed to TVA/mPower): You stated that underground siting provides increased seismic robustness. Do you have any analysis that validates this claim?

Answer 8: Structures below grade typically see an attenuation of ground motion, so amplification of seismic response in the embedded portion of the structure should be, in general, limited to the magnitude of the seismic input of the soil at grade. We are in the process of conducting detailed analyses to confirm and quantify this expected behavior in support of our upcoming license applications.

Question 9 (addressed to TVA/mPower): Please explain the general characteristics of atmospheric release and what issues it may pose for dispersion modeling in remote locations with extreme environmental conditions.

Answer 9: Modeling of potential radioactive releases to the atmosphere accounts for the release location, site-specific meteorological conditions, nearby structures or features that may affect the dispersion, and distance of the receptor from the release point. Potential sites with extreme environmental conditions will be modeled as any other site. Standard plant analysis may not cover ALL possible site locations, and if a particular site is not bounded by the standard plant analysis, then that site may require modification of the standard plant or associated analyses. It is worth noting that when someone refers to "extreme environmental conditions," we typically think of high winds or turbulent conditions. In the context of atmospheric releases, high winds or turbulence is actually a benefit because such conditions actually contribute to increased dispersion.

Question 10 (addressed to TVA/mPower): Many state PUC's and legislators are taking an interest in SMRs. Part of the reason is the small "footprint" (acres) of these SMR design. Can you speak to this issue? How many acres do you need for your plant?

Answer 10: One twin-unit Generation mPower plant requires about 38 acres, which includes the "nuclear island" within the reactor service building, turbine building, and associated security perimeters. The "nuclear island," containing the reactor building and security perimeter is much smaller, less than 7 acres, while the reactor building itself is less than 3 acres.

Most owners will want additional land surrounding their nuclear project to allow for other business requirements, including potential future expansion. Other considerations may also come into play such as emergency planning needs, which will be more fully analyzed as the designs progress. The current Clinch River site has several hundred acres available. This is still a relatively small footprint for the amount of power that can be generated.

Question 11 (addressed to TVA/mPower): The mPower reactors and building is below grade. Probably good for protection against some external hazards but what about flooding risks? And what about fire protection and possible access of fire brigades?

Answer 11: The design and licensing of SMRs will need to take into consideration the particular issues you have identified and answer other questions that will arise because of the innovative approaches being taken. Siting will be the primary barrier to ensure adequate flooding margins are available but engineering features can also be used (as described in more detail in the response to Item 4). Keep in mind that most reactors today are situated below plant grade, so the concern is not unique to SMRs. The requirements for plant staffing will take into account considerations for maintaining sufficient personnel capable of responding and controlling the potential fire risks.

Question 12 (addressed to TVA/mPower): If the SMR design is for two a reactors per site concept, and the refueling cycle is 4yrs. instead of 1 or 2yrs., why is the SMR source term smaller and less need for EPZ?

Answer 12: The question implies a comparison of the total source term of multiple small reactor modules with the source term of a single larger reactor unit. This would only be appropriate if there are accident scenarios that affect multiple modules simultaneously, in which case the same logic would apply to multiple large reactors collocated on a single site.

The fundamental reason for a smaller source term is the significantly reduced size of the core as compared with a GW-scale reactor (i.e., smaller reactors contain less fuel). Further, the overall average burnup is not appreciably different from large reactors and, in the case of the mPower design, the burnup is lower, so the end-of-cycle radionuclide inventory per volume of fuel is also lower.

In addition, opportunities to fully risk-inform certain features to reflect the significant increases in safety margins specific to this design are under evaluation and may be credited if deemed appropriate.

Question 13 (addressed to TVA/mPower): How are SMRs removed from site?

Answer 13: It is not expected that the methods and options for decommissioning the LWR small modular reactors will be different from larger reactors. While the modular nature of the mPower design is a tremendous benefit to the cost of construction, the mPower reactor is not designed to be removed or moved with new or used fuel inserted in the core. Accordingly, decommissioning of the mPower vessel would be performed using traditional techniques. Of course, the mPower reactor does constitute a dramatic reduction in size and amount of material to be decontaminated and disposed of (for example, the entire NSSS is essentially the size of a single steam generator from a current-generation operating plant).

Question 14 (addressed to TVA/mPower): You cited additional questions in new areas for environmental reviews. What are these areas and questions? Do you see a role for a programmatic EIS with tiering on a site specific basis to address?

Answer 14: This is an excellent question and further thinking in this area is warranted. For example, a generic or programmatic EIS approach could handle certain reviews so as to make the site reviews for SMRs proceed more efficiently. Examples of areas that could be addressed are generation alternatives when the total MW's licensed is less than a certain amount, alternative site reviews given the smaller overall impact of SMRs, socioeconomic considerations, etc.

Question 15: Your presentation could have been applied to Solyndra. How is it different? Could it be good money spent on a high risk project at time there is little "good money" available?

Answer 15 (response from DOE): Developing a new technology is an inherently risky endeavor but DOE believes that the SMRs prospects for success are very promising. With the long timelines and large investments needed, it is questionable whether private companies have the wherewithal to see good designs through to commercialization. The DOE program is designed to help accelerate this process, but not bear all of the risk: selected companies will be required to bear at least half of the investment.

Answer 15 (response from TVA/mPower): GmP does not consider the mPower design to be a "high-risk" project. The design is based on evolution of proven technology, rigorous testing, and comprehensive regulatory review. GmP, B&W, and Bechtel are fully committed to carry out this program.

Question 16 (addressed to DOE): What is the status of the Pebble Bed Modular Reactor? Did ESCOM and the U.S. end the project?

Answer 16: The Pebble Bed Modular Reactor is on hold following a decision by the South African government to stop investing to develop the technology. The PBMR had been selected to provide conceptual design work as part of the initial phase of the DOE Next Generation Nuclear Plant program but was not chosen for further development in this effort.

Question 17 (addressed to DOE): Please clarify why DOE has opted for a "4 phase" development & commercialization contrast DOD's technology readiness level (9 steps) used for major weapon acquisition?

Answer 17: The four phases described in the presentation were to emphasize how the technology will need to evolve for SMRs to make a significant contribution to the Nation's energy goals. As a design progresses in the commercialization process the challenges it will need to overcome will change as well. The DOD's 6.X phases reflect a process for technology development that covers similar stages in the evolution with a specific focus on meeting the military's requirements. A key distinction, however, is that the DOD process is intended to result in DOD purchasing the resulting products at the completion of the process. A successful SMR development program will see the technology deployed by power producing companies for the benefits of their customers.

Question 18 (addressed to DOE): Have disposal cost been considered? If not, why?

Answer 18: Disposal costs have been considered and have been assumed to be covered by the current 1-mill per kilowatt-hour fee that is assessed to nuclear plant operators for used fuel management. The costs for disposal were not highlighted because they tend to be quite small in the lifecycle cost estimate of a nuclear plant, especially when compared to the high up-front cost to construct the facility.

Question 19 (addressed to DOE): Why does DOE spend a lot of money on "research" related to life extension of aging reactors while spending this money on SMRs seems to be more appropriate?

Answer 19: DOE believes that to meet our National energy goals the country needs to use all of the clean energy options that are available. The Department clearly believes that SMRs can be an important part of the energy mix and has launched a major program to support their development. The current fleet of nuclear plants provides more than two-thirds of the non-emitting electricity in the U.S. The Light Water Reactor Sustainability R&D program seeks to ensure that these valuable assets can be used safely in the future.

Question 20: Given that each SMR design will need to take advantage of the learning curve, is it realistic to expect there will be enough orders for any given SMR design to take advantage of the phenomena; or will too many SMR designs result in none being successful? Seems like there needs to be at least 10 orders for each design to be successful in lowering costs.

Answer 20 (response from DOE): There is a tension between having a vibrant industry and having the resources for many suppliers to reach commercialization. The DOE program to accelerate the licensing of SMRs intends to provide two awards to strike a balance between consolidating resources on a limited number of designs to maximize the chance of success for each recipient without conferring a monopoly position to a single firm. If the characteristics of SMRs are sufficiently compelling – competitive costs, predictable licensing paths, applicable policy incentives, etc. – then the demand for SMRs may well support many designs. In the end, it is expected that the choices that power companies make in the marketplace will determine the ongoing shape of the industry.

Answer 20 (response from TVA/mPower): The commercialization of SMR technology will benefit from the "learning curve" effect. Because these units are largely modular and factory manufactured, this learning is projected to provide continued improvements in quality and significant reductions in cost for follow-on units. The market will ultimately determine what SMR technology is deployed and to what extent. Higher quality and lower costs will be part of that decision. It is fully anticipated that sufficient SMR orders will exist to maximize the "learning curve" effect and provide a viable long-term market.

Question 21: Will SMRs licensing, R&D, deployment, cost <u>really</u> be less than the small investment of adding scrubbers for small coal units?

Answer 21 (response from DOE): Many analysts have concluded that impending environmental regulations will result in a significant retirement of old coal capacity. Part of the pressures facing owners of coal plants is the uncertainty that future climate policies could require additional investments beyond scrubbers needed to remove pollutants other than carbon dioxide. The expectation is that the R&D and certification costs will be borne as part of the commercialization phase of development. The question for power companies will be whether the costs of building and operating available SMRs will compare favorably to other technologies that could be used to replace aging coal facilities.

Answer 21 (response from TVA/mPower): To be successful, SMR technology must be commercially competitive with other competing technologies for the generation of electric power. There is a wide range of available technologies in the fossil, nuclear, and renewable energy area. Ultimately, adding additional emission controls to existing coal fired units will be evaluated against other new generation options. Because many of the existing coal units are small and old, it may not be financially viable to make the investment necessary to upgrade these units with additional emission control technology.

Question 22 (addressed to DOE): True Risk True Death Re: Dangers of coal, hydro & natural gas vs. nuclear risk. Comment please. Why are annual deaths historical explosions, air pollution, and flooding disasters etc. not considered as a detriment to these nuclear alternatives in a balanced, honest, open non-political manner? e.g. $1.7B by PGE for 2008 explosion

Answer 22: Some of the risks posed by technology options are reflected in the regulations and policies surrounding their use. The civilian nuclear power industry in conjunction with the NRC is at the forefront of assessing the risks of this power source and managing it appropriately. SMRs hold the potential for advancing this record by building upon the lessons of decades of experience and taking advantage of the characteristics of these systems to further improve safety.

Question 23: Will the site evaluation criteria for SMRs be the same as existing nuclear power plants or less strict?

Answer 23: The requirements to be considered for the acceptability of a given site are currently no different than those applied to large nuclear power plants. The manner in which some of those requirements are met by SMRs may vary given the differences in design between SMRs and the larger plants. Ultimately, the NRC will determine whether the requirements and acceptance criteria for SMRs will differ from other plants.

Question 24: What are NRC’s plans for issuing updated regulatory guidance for SMRs?

Answer 24: The NRC staff is preparing design-specific guidance for SMR designs based on design information provided by potential applicants. The Design-Specific Review Standard (SSRS) is similar in structure to the Standard Review Plan (NUREG-0800). The mPower Design-Specific Review Standard Report provides a link to the DSRS sections that have been modified in preparation for the mPower Design Certification Application. These sections marked "Draft for Comment" are available for public comment and stakeholder interaction through this public website and other public meetings including the periodic small modular reactor (SMR) workshops. This is a preliminary comment opportunity. A *Federal Register Notice* (FRN) will be issued in the fall of 2012 for public comment on the mPower DSRS "Draft for Interim Use and Comment". Additional DSRSs will be developed for subsequent designs when design information is provided to the NRC prior to submission of an application. Applicants are not required by current requirements to engage with the NRC in the pre-application activities described herein. The submittals of applicants that choose not to engage the NRC in pre-application activities will be reviewed by the staff using current SRP guidance and methods.

Question 25: To be successful, SMRs need standard regulatory rules & acceptance of a certified design internationally to reduce build costs. How is this to be achieved by regulators internationally? MDEP is a very small start! Comment?

Answer 25: The NRC staff has had preliminary discussions with the Canadian Nuclear Safety Commission regarding sharing of information related to the reviews of the mPower and NuScale SMR designs. If additional countries show interest in reviewing SMR designs, and they are MDEP members, the NRC staff would be interested in participating in discussions with them as we have on other designs. The goal of these MDEP discussions would be to share insights from reviews and work towards common standards, where appropriate. A key concept throughout the work of MDEP, however, is that national regulators retain sovereign authority for all licensing and regulatory decisions.

Question 26: What Regulatory guidance(s) should be used when developing the EIA for SMR sites? What are the major environmental risks for SMRs?

Answer 26: The principal regulatory guidance will be the same as that used for existing sites, i.e. NUREG-1555 and related documents. The NRC staff is developing Interim Staff Guidance which will provide more specific guidance for EISs for SMRs [*sic* NRC assumes the questioner meant EIS]. For the staff's review of SMR licensing applications, early indications are that environmental impacts of alternatives to SMRs, particularly in the areas of energy alternatives and alternative sites, may require new guidance. The NRC staff is working with IAEA on international guidance for EIAs for SMRs.

W21 *- State-of-the-Art Reactor Consequence Analyses*

Session Chair: Kathy Halvey Gibson, RES

Session Coordinator: Jonathan Barr, RES
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The questions below were not answered during the above session:

Question 1: The impact of the dose model used (i.e., LNT or truncated) seems to have a significant impact on the results. What are the next steps planned to remove uncertainty about the choice of dose model?

Answer 1: In estimating health effects from a severe accident, SOARCA calculated the radiation exposure to the population and then applied a dose-response model to analyze early fatality and latent cancer fatality risks. Since the risk of dying of a latent cancer from chronic exposure to low doses of radiation is very uncertain, SOARCA used three implementations of the dose-response model: a linear no-threshold (LNT) approach; exclusion of annual doses less than the U.S. annual dose from natural background and medical exposure (6.2 mSv or 620 mrem) (NCRP 2008); and exclusion of annual doses of less than 0.05 Sv (5 rem) with a lifetime limit 0.1 Sv (10 rem) (based on HPS 2010). These approaches reflect the range of opinions of national and international experts on cancer mortality caused by chronic low-level radiation exposure. The SOARCA project used latent cancer fatality risk factors from a study, entitled “Biological Effects of Ionizing Radiation (BEIR): Health Effects of Exposure to Low Levels of Ionizing Radiation” (BEIR V), the fifth in series of reports published by the National Academy of Sciences, National Research Council.

Although it is impossible to eliminate the uncertainties in the dose-response model, the NRC routinely monitors the development of radiation risk estimates from a wide range of authoritative domestic and international organizations, such as the National Academy of Sciences, National Cancer Institute, and UNSCEAR with the objective of reducing the level of uncertainty in the application of the model to the U.S. population. Risk estimates continually change as new scientific information becomes available and are periodically incorporated into radiation protection recommendations by organizations such as the National Commission on Radiation Protection and Measurements and International Commission on Radiological Protection. NRC evaluates these recommendations for possible incorporation into the technical basis of radiation protection regulations and guidance. Other countries and organizations, such as the International Atomic Energy Agency and European Union, follow a similar regulation-setting procedure.

Question 2: Would it have added value to have some critical non-governmental organization (NGO) peer review?

Answer 2: The SOARCA team assembled a panel of independent, external, technical experts to provide a peer review of the SOARCA project. The members are experts in the fields of risk analysis, severe accident research, emergency preparedness, and radiation health effects. This group reviewed SOARCA's methodology, underlying assumptions, results, and conclusions to ensure that they are technically sound and state-of-the-art. For the same reasons, NRC's Advisory Committee on Reactor Safeguards (a standing group of nuclear safety experts) also reviewed aspects of the project and provided comments. NRC provided opportunity for members of the public, including critical NGOs, to provide feedback on the project. Opportunities included a public review and comment period for the Draft NUREG-1935 report and two public meetings held in February near Peach Bottom and Surry.

Question 3: What is the schedule and planned peer review of the uncertainty analysis?

Answer 3: The SOARCA team presented plans for the uncertainty analysis (UA) to the SOARCA peer review committee (PRC) at two separate meetings. The SOARCA team revised the UA plan to address peer reviewer comments. Subsequently, the PRC was provided an updated list of parameters, distributions, and documentation of technical bases, and a report describing the team's response to individual peer reviewer comments on the UA. This was followed by a teleconference with peer reviewers to close out their comments. In addition, NRC's Advisory Committee on Reactor Safeguards will review the uncertainty analysis and provide feedback to the staff. The UA is planned to be completed in late 2012.

Question 4: Why weren't observations of the peer review combined into a single set of observations? "No consensus was attempted" is not an explanation.

Answer 4: The goal of the peer review committee was to provide an external, independent technical assessment of the project based on a member's specific area(s) of expertise. Observations of the peer review committee members were provided on an individual basis rather than as a consensus report so that all points of view would be expressed.

Question 5: What value of deposition velocity was used in the MACCS2 code? Have you done uncertainty analysis with alternative values of deposition velocity?

Answer 5: Ten different dry deposition velocities are used in SOARCA, each corresponding to an aerosol median diameter size. These are discussed and presented in Section 5.4 of Draft NUREG-1935. In addition, aerosol dry deposition velocity was selected as a parameter for inclusion in the uncertainty analysis. The distributions for dry deposition velocities used in the UA are based on expert elicitation data. The UA will be publicly available when it is completed.

Question 6: How many years of weather data are used to build the meteorological model that generates weather trials?

Answer 6: The SOARCA project used 1 year of meteorological data for the best estimate analysis of each site. This was primarily accomplished through a cooperative effort, with the licensee using onsite meteorological tower observations. Each licensee provided 2 years of weather data. SOARCA based the specific year of data chosen for each reactor based on data recovery (greater than 99% being desirable) and proximity to the target year for SOARCA, which was 2005. Weather data is discussed in section 5.1 and 5.2 of Draft NUREG-1935.

Question 7: You've predicted cancer risks to individuals but populations around most plants have increased. Do those population increases offset calculated risks per capita, producing more fatalities overall?

Answer 7: SOARCA's detailed modeling of emergency planning and response included population data from the 2000 U.S. Census and projected to 2005, which was the target year for SOARCA. Data from the 2010 U.S. Census was not used because most calculations were already completed by the time it was released. Changes in population over the last decade are not expected to have a significant impact on the reported individual cancer fatality risks.

Question 8: Do you know if Fukushima units had obstructions that prevented the ex-vessel core melt from contacting the drywell liner? If not, why didn't this happen?

Answer 8: At the present time, NRC does not have information regarding the configuration of any potential obstructions at Fukushima that may have prevented the ex-vessel core melt from contacting the drywell liner.

Question 9: The Fukushima accident highlighted the potential for large offsite releases of contaminated water. How could this scenario be modeled with current codes?

Answer 9: The MACCS2 (MELCOR Accident Consequence Code System, Version 2) computer code, which was used to calculate offsite consequences in the SOARCA project, does not specifically model offsite releases of contaminated water. Other NRC computer codes are available that evaluate routine releases to rivers and streams and these could be used to assess accident releases to water bodies. Examples include RIVLAK, GROUND, and GRDFLX (see NUREG-0868, "A Collection of Mathematical Models for Dispersion in Surface Water and Groundwater" for more information). The consequence assessment would have to consider situational parameters such as interdiction of public water supplies over some period of time, whether sediment uptake would necessitate longer term restrictions on local use of the water body and restrictions on sport and recreational fishing depending on concentrations and the particular radionuclides involved.

Question 10: For Peach Bottom, was manual post-accident venting of primary containment considered? If so, how and when?

Answer 10: The SOARCA analysis considered post-accident venting through multiple vent pathways, as prescribed in Peach Bottom emergency procedures. It was not credited in the unmitigated scenario calculations, but was fully credited in the analysis of the mitigated scenario.

Question 11: Why did you only report the average values of MACCS2 outputs if you had access to the distributions over weather trials? Why not release all the results now?

Answer 11: As stated within the report, the intent of SOARCA was to produce best estimate calculations of the likely consequences of a severe reactor accident. To meet this objective, the mean results were reported. However, an uncertainty analysis is underway to determine the effect of certain important parameters on the outcomes.

Question 12: Is MACCS2 capable of calculating individual doses for a given geographic region around the plant? If it is, why didn't SOARCA do so? Why just calculate LCF <u>risk</u>?

Answer 12: The MACCS2 computer code is capable of calculating absolute numbers of latent cancer fatalities (LCF) for SOARCA scenarios; however characterizing health consequences in terms of numbers of LCFs isn't very meaningful because there is considerable uncertainty when attributing very small exposures across a large number of individuals. Therefore, SOARCA's calculations of health consequences are reported in terms of the individual risks of an early fatality and a long-term cancer fatality. The presentation of health effects in terms of individual average risks enables a more direct comparison to the overall individual annual risk of a cancer death from all sources in the U.S. and comparison to the NRC Safety Goal to provide context to the numbers reported.

TECHNICAL SESSIONS
Thursday, March 15, 2012, 8:30 a.m. 10:00 a.m.

TH22 *- Fire Protection- Regulatory and Industry Perspectives on NFPA 805 and Circuit Analysis*

Session Chair: Alex Klein, NRR

Session Coordinator: Alayna Pearson, NRR
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The questions below were not answered during the above session:

Question 1: What regulatory regulation (ie: process) would be applied by licensees to maintain the fire protection licensing PRA basis resulting from the application of NFPA 805? How are the fire protection licensing basis changes managed and controlled by processes that maintain the safety envelope for nuclear operations?

Answer 1: The Fire PRA used in support of 10 CFR 50.48(c) must be maintained consistent with the referenced National Fire Protection Association (NFPA) 805 Standard. The Standard requires that the PRA represent the as-built, as-operated and maintained plant. Further guidance is provided in Regulatory Guide (RG) 1.205, including the template license condition that provides the conditions under which licensees need to make a new license application to the NRC for plant changes to ensure the fire protection safety envelope is properly addressed and maintained. In addition, RG 1.205 references RG 1.200 (which in turn endorses with clarifications the ASME/ANS PRA Standard), in which the staff position and guidance is provided on how PRA technical adequacy is addressed and maintained consistent with this application.

Question 2: Why would a licensee ask for a confirmatory order and risk a NOV against the confirmatory order?

Answer 2: SECY-12-0031 informed the Commission of the staff's available alternatives in addressing licensees who will not submit their LAR on their docketed scheduled date.

Voluntary transition to NFPA 805 is viewed by the NRC as an acceptable way to address multiple, longstanding non-compliances with the existing deterministic fire protection requirements. The NRC is facilitating the adoption of NFPA 805 by granting enforcement discretion for most non-compliances identified during the transition activities. This encourages licensees to find, report, and correct legacy fire protection design issues. Plant safety is being maintained during this period through the use of compensatory measures.

Some licensees have missed their scheduled LAR submittal date and lost enforcement discretion. The delay has caused the staff to consider appropriate responses for failure to submit an acceptable LAR as scheduled. In considering alternatives, the staff identified three general objectives: (1) establish a framework for effective regulatory response that is consistent with the existing Enforcement Policy; (2) ensure the timely completion of committed licensee actions to transition to NFPA 805; and, (3) reinforce public confidence by taking steps that address legacy fire protection non-compliances, rather than continued reliance on long-term compensatory measures.

To gain additional time under enforcement discretion for a licensee to complete their LAR submittal, the NRC may allow the licensee to swap with another site or sign a confirmatory order, identifying a new submittal date. Signing a CO with continued enforcement discretion provides the opportunity for completion of the LAR and contributes to public confidence by illustrating that the licensee is actively managing their transition to adopt NFPA 805.

The staff determined that these alternatives were most effective in addressing the overall objectives as outlined above.

Question 3: NRC Stated that they would pursue the technical and safety details of cable issues MSO/SSO… Does this mean that the NRC is no longer considering plants' licensing basis as valid? Have they done or will they do? For plants that are SSO based on their licensing basis is the NRC saying they must change to MSO without going through back fit? If a plant is committed to the NEI MSO guidance and is making progress but will not be done because of outage modification, by Nov 2012- Is the NRC going to issue a violation?

Answer 3: Regulatory Guide 1.189, Revision 2, provides guidance regarding the treatment of circuit failures, it does not redefine a plant's licensing basis. Where a plant's licensing basis is lacking clarity with regard to the treatment of spurious actuations, the NRC staff will rely on current guidance in Regulatory Guide 1.189, Revision 2. This will be addressed on a case by case basis and will be informed by the information in the plant's licensing basis. If the staff determines that a plant has a licensing basis that is not consistent with NRC guidance, then the NRC may pursue a backfit.

Plants that are not done implementing multiple spurious actuation guidance by November 2012, will not have enforcement discretion per Enforcement Guidance Memorandum 09-002. In this case the Reactor Oversight Process will be applied on a case by case basis with consideration of the factors in Inspection Manual Chapter 9900 for timeliness of corrective actions.

Question 4: At the Nov 2011 public meeting, it appeared NRC was increasing LAR scope (seismic standpipes, NEI 00-01 gap analysis, etc). Why is this? Also, why do we have to address FSAR changes in the LAR now when this is driven by the regulation?

Answer 4: With respect to "seismic" standpipes, NFPA 805 Section 3.6.4 requires provisions for manual fire fighting post-safe shutdown earthquake (SSE). Several licensees have only addressed the licensing status of their standpipes without addressing the ability to meet the standard requirement. Even though a licensee may predate the requirement for seismic standpipes and hose stations, they still need to address the NFPA 805 requirement on their ability to manually fight fires in areas containing systems and components needed to perform the nuclear safety functions in the event of an SSE.

With respect to NEI 00-01, revision number; several licensees have referenced NEI 00-01, rev. 1 in their submittal, while the RG 1.205 endorses NEI 00-01, rev. 2. In order to avoid the necessity of a generic RAI for remaining licensees, it is necessary to provide a gap analysis with a technical justification that demonstrates that use of NEI 00-01 rev. 1 does not adversely impact the analysis.

With respect to the FSAR, in accordance with Section 4.6.1 of the current industry guidance document (NEI 04-02, Revision 2), the LAR should include a discussion of the changes to the FSAR necessitated by the transition and a statement that the changes will be made in accordance with 10 CFR 50.71(e).

Question 5: While NFPA 805 is focused on changes to operating plants, how do you foresee it affecting plants under construction? Also, how would these 805 insights inform licensing of new designs and plants?

Answer 5: Reactors licensed under Part 52 (New Reactors) and those under construction that will be licensed under Part 50; can benefit from a performance-based fire protection review to locate risk-significant locations that need strategies to reduce the possibility of core damage for fire.

NFPA 805 was developed for existing reactors that are licensed under 10 CFR 50. The NFPA Technical Committee on Fire Protection for Nuclear Facilities developed NFPA 806, "Performance-Based Standard for Fire Protection for Advanced Nuclear Reactor Electric Generating Plants Change Process" for reactors licensed under 10 CFR 52. However, at this time, the NRC has not endorsed NFPA 806 as an acceptable licensing basis.

Question 6: How does a licensee address beyond design basis events (including design basis threat), where barriers may be defeated or unavailable, and involve multiple fire areas?

Answer 6: Order EA-12-049 was issued in March of 2012 and requires licensees to develop a three-phase approach for mitigating beyond-design-basis external events. Guidance and strategies required by the Order would be available if the loss of power, motive force, and normal access to the ultimate heat sink to prevent fuel damage in the reactor and spent fuel pool, affected all units at a site simultaneously. Additional details on an acceptable approach for complying with the Order will be contained in a final Interim Staff Guidance (ISG) document scheduled to be issued in August, 2012.

***TH23** – License Renewal – Perspectives on Current and Subsequent License Renewal, Part 1 of 2 (Double Session with TH32)*

Session Chair: Melanie Galloway, NRR

Session Coordinator: Albert Wong, NRR
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The questions below were not answered during the above session:

Question 1 (addressed to Bo Pham, NRC): Qualification of Equipment; how is safety qualified equipment managed with regard to aging? For instance, electrical penetrations within containment or safety reactor instrumentation for license renewal between 40 and 60 years.

Answer 1: NRC regulation (10 CFR 50.49) requires a licensee to establish a program to qualify electrical equipment such that when it reaches its end of life condition it will meet its design requirements during and following a design basis accident for the environmental conditions at the equipment's location. Among the environmental conditions considered are those resulting from loss of coolant accidents (LOCA), high energy line breaks, and post-LOCA radiation.

Under this program, equipment is environmentally qualified by testing to its end of life condition or its condition at the end of the operating term (e.g., 40 years). Environmentally qualified components with a qualified life of at least 40 years are within the scope of license renewal. As stated in the Generic Aging Lessons Learned (GALL) Report, for the purposes of license renewal, environmental qualification (EQ) programs that implement the requirements of 10 CFR 50.49 are considered an acceptable aging management program (AMP) for license renewal under 10 CFR 54.21(c)(1)(iii). Environmentally qualified equipment in scope of license renewal is reanalyzed per 10 CFR 50.49(e) for the period of extended operation (i.e., up to 60 years). The reanalysis is performed under the applicant's existing EQ program.

For components with a qualified life of less than 40 years, the requirements of 10 CFR 50.49 continue to apply with equipment either refurbished, replaced or reanalyzed prior to reaching the components' design life (end of life condition) throughout the period of extended operation. Therefore, the applicant's EQ program continues to manage the environmental qualification of electrical components important to safety for the license renewal period of extended operation (40 to 60 years).

Question 2 (addressed to Melanie Galloway, NRC): Does NRC measure the influence of aging on a global level of risk through Probabilistic Risk Assessment (PRA) Level 1 or Level 2 internal events and hazards?

Answer 2: The amended 1995 License Renewal Rule in 10 CFR Part 54 is not a risk-informed rule in that risk is not used to manage the adverse effects of aging on the passive structures and components in 10 CFR 54.21(a)(1)(i). The influence of aging on components is controlled and/or managed through maintenance, testing, inspection and replacement programs. These programs ensure that the effects of aging do not impact component failure rates.

NRC has sponsored research programs to examine how aging could be factored explicitly into the PRA models. The research has not provided a practical way to explicitly incorporate aging into the PRA models.

Question 3: What changes are expected to the Reactor Oversight Process (ROP) during the Period of Extended Operation (PEO)? And for subsequent license renewal?

Answer 3: The current baseline inspections and the performance indicators in the ROP are under constant review for improvements and enhancements to ensure plant performance continues to be adequately monitored during the first license period as well as during the PEO. For example, Inspection Procedures on Flood Protection Measures (ML11244A012) and Problem Identification and Resolution (ML112360542) have recently been expanded to include aging management inspections. In addition, work is underway to enhance other baseline procedures for aging management purposes as well.

With respect to subsequent license renewal, if and when additional inspection criteria for license renewal past 60 years are identified, they will be used to evaluate the adequacy of ROP for assessing the plant performance. Changes will be made to enhance the ROP if gaps are identified.

Question 4: What efforts are underway in consensus codes and standards bodies, such as, ASME with respect to developing rules for Aging Management Programs? What international codes and standards are available? How do these affect the licensing basis?

Answer 4: There are several activities in the codes and standards areas that relate to aging management programs. Each of the codes and standards organizations has its own areas of responsibility, and the NRC utilizes the products of these organizations (e.g., code or standards) in the GALL Report to the extent possible. In some cases, none of these organizations addresses certain areas that require aging management, and in those cases the GALL Report provides the basis for the development of AMPs.

The Nuclear Energy Standards Coordination Collaborative (NESCC) is a joint initiative of the American National Standards Institute (ANSI) and the National Institute for Standards and Technology (NIST) to facilitate and coordinate the timely identification, development, and revision of standards for the design, operation, development, licensing, and deployment of nuclear power plants. Other participants in NESCC include the NRC, ASTM International, the American Society of Mechanical Engineers (ASME), the American Concrete Institute (ACI), and the National Fire Protection Association (NFPA). NESCC includes task groups on concrete repair, cable aging, and high density polyethylene piping (HDPE).

From an international perspective, the International Atomic Energy Agency (IAEA) is developing an International GALL Report, or IGALL Report, which provides an international perspective on aging management needs for extended plant operation, considering not just U.S.-style boiling water reactors (BWRs) and pressurized water reactors (PWRs), but also Russian-style PWRs and pressurized heavy water reactors, including CANDU reactors (“CANada Deuterium Uranium"). Although IGALL has used the U.S. GALL as a starting point, the NRC is assessing information from other participants to determine if there are insights that can be used to enhance the GALL Report for use by U.S. licensees.

The IGALL Report should be issued by early 2014. The codes and standards and international activities do not impact the licensing basis for U.S. plants, except as explicitly identified in the NRC’s regulations.

Question 5: Are Aging Management Programs NRC’s requirements or licensee self imposed requirements? Explain how these programs are enforced by NRC regulations.

Answer 5: The GALL Report (NUREG-1801) is a regulatory guidance document that represents one acceptable way to manage aging effects for license renewal, but applicants may also propose plant-specific alternatives for staff review in their applications. Use of aging management programs recommended by the GALL Report is not required, but its use facilitates both the preparation of a license renewal application by an applicant and timely review by NRC staff. Aging management programs are required by the Commission’s regulations to demonstrate that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the current licensing basis during the period of extended operation. The Commission’s regulations require license renewal applicants to submit an FSAR supplement containing descriptions of aging management programs. That FSAR supplement become parts of the licensee’s FSAR upon issuance of the renewed license, and thus changes to AMPS are subject to the change control process in 10 CFR 50.59.

TH24 *- New Reactors Licensed under 10 CFR Part 52: The End Game – Coordinating Rules, Licenses, and Mandatory Hearings*

Session Chair: David Matthews, NRO

Session Coordinator: Donna Williams, NRO
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Questions submitted during the above session were answered during the session’s Q/A period.

TH-25 *Post-Fukushima International Research*

Session Chair: Brian Sheron, RES

Session Coordinator: Leroy Hardin, RES
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The questions below were not answered during the above session:

Question 1 (addressed to Brian Sheron, NRC): With the sudden emphasis of costs associated with consequences, such as loss of use of assets, evacuation, mental anguish, etc., has there been any renewed effort to assess these costs from accident events?

Answer 1: The NRC has always considered off-site impacts in its regulatory analyses and its guidance is that the impacts should be quantified whenever possible. ("Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," NUREG/BR-0058, Revision 4). Examples of such costs include among other things loss of use of property and evacuation costs. The NRC is in the process of revising/updating values used in its regulatory analyses.

Question 2 (addressed to Brian Sheron, NRC): For Level 3 PRA at Vogtle, how much of the input data will be considered proprietary Southern Company information and therefore not part of the public record?

Answer 2: The NRC is in the early stages of interacting with Southern Nuclear regarding access to information that would support the Level 3 PRA for Vogtle, Units 1 and 2. Therefore it is too soon to tell what information can or cannot be made available to the public. We do expect that there will be a certain amount of information used for the Level 3 PRA project that will be commercial proprietary or otherwise non-public, due to its sensitive nature (e.g., engineering design calculations, plant procedures). Although detailed information supporting the PRA project may be subject to withholding, the NRC does intend to publish a final publically available report that will provide background information on the study and summarize study results. Additionally, the NRC intends to hold periodic public meetings after completing key project milestones to share information regarding the study with external stakeholders.

Question 3 (addressed to Brian Sheron, NRC): What is the current NRC thinking regarding early containment venting?

Answer 3: See question 5 under Questions for Brian Sheron and Masashi Hirano.

Question 4 (addressed to Brian Sheron, NRC): Does the industry plan any research on the environmental impacts (long term property damage, infrastructure, social, long term personnel displacement, etc.)? If so, who is conducting this research?

Answer 4: Questions related to what research the industry is or is not performing are better directed to the Nuclear Energy Institute.

Question 5 (addressed to Brian Sheron, NRC, and Masashi Hirano, JNES): Is experience of Fukushima accident enough to recognize necessity of installing filter venting system? If so, why do we need additional research to convince the necessity?

Answer 5 (response from Masashi Hirano, JNES): In Japan, in my understanding, the industry is proposing to install filtered venting system. I assume some R&D activities would be done by the industry to newly design it or to select suitable design from existing ones. From regulatory viewpoint, I don't know at the moment whether we need additional research in this area.

Answer 5 (response from Brian Sheron, NRC): The NRC's near term review ("Recommendations for the Enhancing Reactor Safety in the 21st Century: The Near-term Task Force Review of Insights from the Fukushima Dai-ichi Accident," July 12, 2011, Nuclear Regulatory Commission) recommended making those venting capabilities a regulatory requirement for Mark I and II containments, and evaluating the need for venting for other containment designs. The NRC staff recently issued orders on March 12, 2012 implementing that recommendation, which will lead to hardened, reliable vents in all Mark I and Mark II containments.

In July 2012, the staff is expected to respond to the Commission's request for a decision paper on requiring filtered containment venting systems for Mark I Mark II, and a plan for evaluating venting for other containment designs. The staff's paper will address the various containment venting issues, and provide recommendations to the Commission at that time.

Question 6 (addressed to Jacques Repussard, IRSN, and Javier Reig, OECD/NEA): What is the relationship between the safety initiatives that you described and the trade harmonization safety directives (e.g., the Nuclear Safety Directive enacted in 2008/9)? In particular, please comment on the relationship of hydrogen generation with the ATEX Directive (explosive atmospheres), if it applies.

Answer 6 (response from Jacques Repussard, IRSN): The « stress tests » which are being carried out in the EU result from a concerted political initiative at the level of the top governing body of the EU, the Council of Ministers. The political backdrop of this initiative is that of a progressive policy to harmonize nuclear safety practice in the EU, which has been embodied so far by the adoption of two "directives," one on nuclear safety and one on management of radioactive waste. There are also a number of harmonized regulations in the field of radiation protection standards. The ATEX directive applies to electrical equipment to be used in explosive atmospheres, in order to allow their marketing across the EU. It does not address the issue of limiting or preventing directly the risks resulting from such explosive atmospheres in industry plants, nuclear or other.

Answer 6 (response from Javier Reig, OECD/NEA): Not very familiar with the ATEX directive but I think it applies to different industry: petrochemical and mining. Objective in NPPs is to avoid damage of hydrogen explosion to containment function.

Question 7: The Fukushima accident included large offsite releases of contaminated water. It seems that this is not included in severe accident consequence analysis. Are there any plans to address this gap?

Answer 7 (response from Masashi Hirano, JNES): The IAEA Safety Guide NS-G-2.15 on the accident minaret (AM) measures defines the objectives of the AM measures as (a) to prevent the escalation of the event into a severe accident; (b) to mitigate the consequences of a severe accident; (c) to achieve a long term safe stable state. The point raised by this question belongs to (c) which, I believe, has not been widely discussed so far. We need to address this issue in the international community because it may be difficult for any single country to prepare counter measures for this.

Answer 7 (response from Frank-Peter Weiss, GRS (GER): There are no current or planned activities at the GRS to address this point by calculations. Nevertheless, this modeling gap has certainly to be closed in the further development.

Answer 7 (response from Jacques Repussard, IRSN): In France, the possibility of large ground contamination in case of Gen II PWR severe accident leading to basemat penetration is considered and discussed at the occasion of 10-yearly safety reviews. In the frame of post Fukushima action plans the investigations efforts have been intensified in two directions:

- Reinforcement of the basemat and/or improvement of the efficiency of corium cooling during MCCI phase,

- Consideration of implementation of a "barrier" to protect the ground water below the containment (to be installed in prevention, not during an accident).

These two topics are part of possible Gen II PWRs upgrade in the context of LTO.

Answer 7 (response from Javier Reig, OECD/NEA): (response to questions 7, 11, 12 and 14): The Fukushima accident will have an impact on future safety research, in particular to see how can be improved the knowledge and the response to severe accidents. Even if the scenarios are not completely new, there are some elements in Fukushima accident which need to be revisited (i.e. longer loss of safety functions). The decision on what research needs to be done will come from the countries interested and based on their regulatory policies, so not only based on probability approach.

Answer 7 (response from Brian Sheron, NRC): The MACCS2 (MELCOR Accident Consequence Code System, Version 2) computer code, which was used to calculate offsite consequences in the SOARCA project, does not specifically model offsite releases of contaminated water. Other NRC computer codes are available that evaluate routine releases to rivers and streams and these could be used to assess accident releases to water bodies.

Examples include RIVLAK, GROUND, and GRDFLX (see NUREG-0868, “A Collection of Mathematical Models for Dispersion in Surface Water and Groundwater” for more information). The consequence assessment would have to consider situational parameters such as interdiction of public water supplies over some period of time, whether sediment uptake would necessitate longer term restrictions on local use of the water body and restrictions on sport and recreational fishing depending on concentrations and the particular radionuclides involved.

Question 8: What are the design requirements for containment hardened vent piping? E.g., if existing piping is not missile protected, will upgrade be required?

Answer 8 (response from Frank-Peter Weiss, GRS (GER): As a consequence of the severe accidents at TMI and Chernobyl recommendations were published in the late 80s by the reactor safety commission (RSK). It contained amongst other severe accident management measures the backfitting of a filtered venting system in the German light water reactors. Important aspects cover the design and set-points for operation, the loads to be considered and construction requirements (see also EU Stress Test, National Report of Germany).

The filtering of the released containment atmosphere is achieved by either a combination of a variable-pressure venturi gas scrubber for the retention of iodine and a metallic fiber fleece filter for the retention of aerosols, or a combination of a metallic fiber fleece filter for the retention of aerosols and a molecular sieve for the retention of iodine (KTA-GS-66). In most German plants the systems are installed inside buildings which are protected against some external hazards. Nevertheless a systematic protection of the systems against external hazards was not requested as it was not assumed that a severe accident in the core may be the direct consequence of an external event (beyond such NPP protection).

Answer 8 (response from Jacques Repussard, IRSN): Design requirements for hardened safety core, including containment filtration and venting system will be proposed by the licensee in June 2012. It will be analysed by IRSN for the end of 2012. It is then too early to answer the question. From IRSN point of view, the containment venting system should be reinforced to withstand beyond design earthquake (the EQ level to be considered being defined site by site). The question of the feasibility may nevertheless be raised by EDF.

Safety requirements aim that an EQ should not prevent the operation of safety-related equipment (in particular, the earthquake resistance of the structures on which the filter is installed must be maintained).

Answer 8 (response from Brian Sheron, NRC): The NRC staff issued Order EA-12-050, “Order Modifying Licenses with Regard to Reliable Hardened Containment Vents” on March 12, 2012 (ADAMS Accession No. ML12054A696). The performance objectives, design features, quality standards, and the programmatic Requirements for the reliable hardened vent system are delineated in Attachment 2 to the order. Additional details for complying with the Order will be provided in an Interim Staff Guidance (ISG) that is scheduled to be issued in August 2012, after due participation from the stake holders in the development of the guidance.

Question 9: My main conclusion on Fukushima is: the protection against a known risk failed. There was something like a blind spot of attention. But why? Does anybody on the panel address this topic with the goal to prevent blind spots regarding known risks?

Answer 9 (response from Masashi Hirano, JNES): I think this is one of the most important lessons learned since this is relevant to safety culture. This is also relevant to how to be prepared for low frequency and high consequence hazards. If people think its frequency is low, it tends to be in a blind spot of attention, even if its uncertainty is large. We need to continue to discuss how not to be complacent but to be vigilant on these hazards.

Answer 9 (response from Frank-Peter Weiss, GRS (GER): Germany puts a lot of effort to exclude blind spots regarding known risks. Therefore Germany sets a high value on the independence of different institutions and organizations taking part in the nuclear licensing and supervising procedure. The GRS as the main technical support organization in nuclear safety for the German federal government is independent of any political and commercial influences and a non-profit organization which underlies special requirements (principles for TSOs) exclusively dedicated to nuclear safety. The interaction of different approaches and means (e.g. deterministic and probabilistic methods, precursor analyses, evaluation of operational experience feedback) aims to assure a well-balanced safety concept in order to exclude single contributors dominating the overall risk.

Further, the complementary safety assessment in Germany and the investigations in the framework of the European Stress Test especially address the identification of weak points and insufficient robustness. For example, the NPPs in Germany are designed against earthquakes likely to appear with a frequency of > 10^{-5} /a (and in any case against earthquakes with a level of intensity equal to VI). In case of impacts caused by flooding, the German NPPs are designed with respect to a plant-specific 10,000-yearly flood. Other natural hazards are covered also by the consideration of postulates (wide set of events), for example addressed within the scope of safety analyses. Protection against man-made hazards shows in the case of a postulated aircraft crash that all German NPPs in operation are able to withstand the load assumptions (and the related requirements) resulting from the military aircraft of the type "Phantom" or medium commercial aircrafts. The German NPPs in operation also withstand blast waves with a maximum excess pressure of 0.45 bar. The entry of explosive materials is prevented and the existence of toxic gases is considered at all plants in a plant-specific way.

Answer 9 (response from Jacques Repussard, IRSN): Nuclear industry is man-made, therefore subject to potential failure. The idea is not so much to identify all possible "blind spots," which seems unfeasible, but to anticipate the probable rare occurrence of such unanticipated events, by reinforcing the generic capability of NPPs to resist to situations potentially leading to the destruction of all barriers containing radioelements, in accordance with the basis of defense in depth. The European approach therefore considers that "beyond design accident conditions" should also be considered, to ensure as far as possible reasonably that their occurrence will not lead to large environmental releases.

Answer 9 (response from Javier Reig, OECD/NEA): The NEA working groups are looking at operating experience to identify occurred events outside the design basis which could be precursors for a severe accident, and so be able to prevent it.

Answer 9 (response from Brian Sheron, NRC): The NRC's intended strategic outcome (see http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1542/v17/s1/sr1542v17s1.pdf) is to prevent the occurrence of any reactor accidents. The NRC licenses reactor plant designs to meet regulations that ensure adequate protection against known risks, including external hazards such as earthquakes and flooding. Since the accident at Fukushima, the international nuclear community including the NRC, have been evaluating what can be learned from the accident regarding the adequacy of current requirements and to what extent plants should be prepared for external hazards beyond those the plant is designed to withstand.

Through our assessment of plant operation, the NRC has identified certain issues involving public health and safety, the common defense and security, or the environment that could affect multiple entities under NRC jurisdiction. We document and track resolution of these generic issues (GIs). The generic issues program (GIP) includes five distinct stages that may be exercised: Identification, Acceptance Review, Screening, Safety/Risk Assessment, and Regulatory Assessment. In addition, the GIP tracks and reports the GI status and resolutions to Congress and the public annually. The resolution of these issues may involve new or revised rules, new or revised guidance, or revised interpretation of rules or guidance that affect nuclear power plant licensees, nuclear material certificate holders, or holders of other regulatory approvals.

A recent example of an area that we are now closely studying is GI-204, which is titled "Flooding of Nuclear Power Plant Sites Following Upstream Dam Failure". The evaluation was initiated following recent NRC staff findings under the Reactor Oversight Process with respect to flooding protection which could be challenged by potential upstream dam failure. The screening assessment of the generic issue concluded that further evaluation of external flooding of nuclear power plants due to an upstream dam failure was warranted, which required a risk-informed evaluation of the impact of potential flooding scenarios, such as the likelihood of potential dam failures, flooding analysis, and consequential impacts at nuclear power plants. No immediate safety concerns were identified during the conduct of the screening assessment.

Question 10: What is being done to ensure that emerging nuclear countries will have strong regulatory capacities and procedures? What controls are in place to ensure that they will proceed with adequate focus on safety?

Answer 10 (response from Frank-Peter Weiss, GRS (GER): The use of nuclear power falls to national sovereignty. Nevertheless, the IAEA take care about the global enhancement of nuclear safety ("to provide a strong, sustainable and visible global nuclear safety and security framework […]"). In case of member states interested in nuclear power, IAEA especially provides technical assistance, missions, workshops and new/updated technical publications. Further programs and networks also aim to achieve an adequate level of safety in emerging nuclear countries: Through cooperation under the Instrument for Nuclear Safety Co-operation (INSC), the EU supports non-EU countries ensuring that nuclear safety is the leading concept in regulatory and legal frameworks and operational decisions. GRS is one of the major contributors to the implementation of the INSC projects. The European Technical Safety Organisations Network (ETSON) also contributes to foster the convergence of technical nuclear safety practices within the European Union and beyond.

Answer 10 (response from Jacques Repussard, IRSN): The IAEA has set up a number of initiatives to help "new entrants" to deal correctly with such issues, particularly to promote the implementation of IAEA standards and guides. In addition, the EU, as well as the USA, has launched and funded an active cooperation program intended to support the development of the appropriate resources in such countries. Considering the international consensus to respect strictly the national character of responsibilities for nuclear safety, the only approach to some degree of control resides in the promotion of peer review mechanisms, and of transparency to all stakeholders, line with best practice in IAEA member states.

Answer 10 (response from Javier Reig, OECD/NEA): The IAEA has a programme to ensure that new entrant countries have adequate regulatory framework and authority, but not only, also to ensure that the operator is technically qualified and its organization is fully capable. The NEA is supporting the IAEA and a group of selected countries with the material produced by its working groups.

Answer 10 (response from Brian Sheron, NRC): Each country which considers developing a nuclear power option is responsible for ensuring the safe operation of that program and its effective regulation. To demonstrate their adherence to the highest principles of nuclear safety, countries are encouraged to become parties to the major international legal instruments such as the Convention on Nuclear Safety, which establishes binding obligations relating to the safe use of nuclear energy.

In order to assist countries as they make their energy plans, there are a number of options, some multilaterally-based, and others through bilateral channels. The International Atomic Energy Agency (IAEA) provides many avenues to support the development of national nuclear programs. Through the IAEA, Member States may be requested to provide support, such as participation in assessment missions which identify strengths and weaknesses.

The country seeking a nuclear power program can also develop bilateral relationships with other countries and their national regulatory bodies to support their efforts. When requested, the U.S. Government and the Nuclear Regulatory Commission have provided a range of assistance activities.

Question 11: In the past, resources spent on severe accident R&D took into account the low probability of severe accidents. Currently, it appears that resources for severe accident R&D ignore the low probability of a severe accident. Does this reflect a change in R&D priority which may reduce resources to other R&D topics?

Answer 11 (response from Masashi Hirano, JNES): My point is that R&D activities are important to maintain technical competence and expertise. I believe the severe accident R&D should have been kept at a certain level. It may be dangerous to prioritize the R&D activities based only on risks because sometimes their uncertainties are very large. We need to take a more holistic approach.

Answer 11 (response from Frank-Peter Weiss, GRS (GER): In Germany there was basically a continuous funding of R&D activities dedicated to severe accident in the past 2-3 decades. This has not been changed, but emphasis is given to the phenomena and the event sequence of the Fukushima accidents. This addresses for example the behavior in the spent fuel pools in case of emergency. Further, the transfer of Fukushima lessons learnt to German NPPs is evaluated and the range of application of simulation codes is extended in order to cover further beyond design basis scenarios. However, a reduction of resources to other R&D topics cannot be stated.

Answer 11 (response from Jacques Repussard, IRSN): Not all countries have treated severe accident R&D according to the probability of accident. Considering the huge potential consequence of such an accident, in at least in environmental and economic terms, IRSN in particular has since TMI spent considerable resources on R&D in severe accidents, and will continue to do so. Other fields also need to be addressed, of course, and this should be done in cooperation internationally, to use scarce resources as best as possible. This is the reason why NEA/CSNI exists, inter alia, offering the international benefit of the joint projects approach.

Answer 11 (response from Brian Sheron, NRC): Severe accident research has been the subject of NRC and industry research for many years, especially since the Three Mile Island accident in 1979. The NRC continues to evaluate and act on the lessons learned from the March 2011 nuclear accident in Japan to ensure that appropriate safety enhancements are implemented at nuclear power plants here in the U.S. This emphasis has led to some redirection of Agency resources, including research, toward severe accidents. The significant potential consequences of severe accidents warrant an adequate understanding of them, so the Commission has directed resources accordingly to address questions raised by the Fukushima accident. It should also be noted that severe accidents have actually occurred, and assessment of their probability is not a precise science. So the Agency conducts a prudent and, in our view, appropriate amount of research in the area. This emphasis on severe accident research is also consistent with Congressional direction on response to the Fukushima accident.

However, NRC-sponsored research continues on many subjects not directly linked to severe accidents. The Commission decides on an ongoing basis the appropriate use of Agency resources. Such decisions are informed in part by assessments of risk (probability of potentially undesirable events and outcomes, and the consequences of their occurrence).

Question 12: Five decades of safety research have not helped to prevent the Fukushima accident. On the other hand, the accident progression did not offer too many surprises. Hence, the answer should not be: more of the same, but: Which other way should we go?

Answer 12 (response from Masashi Hirano, JNES): I believe it was a fatal mistake that we have not taken into account any external events in applying PRA for developing and evaluating the accident management measures. I believe we have basically enough knowledge on severe accident phenomena and therefore, future research should be directed to synthesis of such knowledge and its application to prevention and mitigation of the consequences of severe accidents.

Answer 12 (response from Frank-Peter Weiss, GRS (GER): Investigations are and will be performed to get deeper insights in the progression of the accidents and to evaluate the transfer of the lessons learnt to German NPPs. To propose changes in order to enhance safety, it is necessary to consider the findings of the current investigations. Nevertheless, the occurrence of the Fukushima accident was minor a factor of safety research. Rather, it is important to handle and assess safety issues independently from political and commercial influences and to regard external events in a systematic way.

Moreover, to my opinion, the approach followed in the German and the European stress tests, namely to look at the robustness against cliff edge effects is the right way to go, but certainly needs some refinement in the future.

Answer 12 (response from Jacques Repussard, IRSN): First, too much detailed information on the accident sequences in the Fukushima reactors is missing today to be able to state that the accident progression could be exactly forecast by existing models. Second, it seems clear that knowledge needs to be further elaborated on issues such as hydrogen accumulation phenomena, containment venting mechanisms, or spent fuel pool potential accidents and their prevention. Third, a clear way forward to enhance safety seems to reinforce defence in order to strengthen reactor systems' resistance to situations of prolonged black outs and / or loss of heat sink, and to ensure efficient accident management capabilities on site as well as off site. This is the objective of the so called "hardened safety cores" which are being proposed for French power reactors following the "stress tests" (also called in France complementary safety evaluations).

Answer 12 (response from Brian Sheron, NRC): The safety research activities have indeed provided insights and produced simulation tools and accident procedures designed to prevent or mitigate the consequences of severe accidents. Fukushima led to a worldwide review of the potential for and response to severe accidents. The evaluation of the accident at Fukuskima has already and will continue to yield useful information for severe accident research.

Question 13: Is there a concerted effort to minimize duplicated severe accident R&D amongst the different international organizations?

Answer 13 (response from Frank-Peter Weiss, GRS (GER): It is important to mention, that there are national coordinated meetings to agree the goals and the content of reactor safety research between the institutes involved, to discuss potentials for collaboration. This is done within the German competence alliance on nuclear technology.

On the international level, a coordination of research activities is realized within the European Sustainable Nuclear Energy Technology Platform (SNE-TP). As an example, responding to the last EURATOM call, the project "Code for European Severe Accident Management (CESAM)" was proposed. It deals explicitly with the impact of the nuclear accident in Japan on severe accident management and is coordinated amongst 8 partners. Further networks like ETSON also aim at contributing to the definition of nuclear safety research and development programs and to elaborate a common position on safety-research priorities. Moreover, NEA/CSNI spends a lot of efforts to the coordination of severe accident research in the OECD countries and beyond.

Answer 13 (response from Javier Reig, OECD/NEA): The NEA members are all members of the IAEA, so it is in their interest to avoid this duplication. Traditionally the NEA has provided an effective framework for performing safety research and there is no reason for changing that approach. The IAEA management supports this view.

Answer 13 (response from Brian Sheron, NRC): Yes, NRC is actively working with its counterparts abroad, on a bilateral and multilateral basis, to discuss current and planned severe accident research programs. The OECD-NEA is one venue to discuss this area of work and minimize duplication of effort to the extent possible. The NRC's Cooperative Severe Accidents Research Program (CSARP), which is centered on the NRC's MELCOR code, is another venue where experts from several countries will meet and can benefit from each other's experience in severe accident research.

Question 14: In light of the Fukushima accident, what long term research is being conducted to reduce or eliminate some challenges, e.g., non zircaloy cladding to limit H_2 releases.

Answer 14 (response from Jacques Repussard, IRSN): R&D efforts on new fuel designs, beyond the optimization of currently licensed fuel systems for light water reactors, are related to generation IV reactors. The lead time to operational applications of such innovative systems is such that it is indeed more compatible with GEN IV projects than with the current reactor fleets.

Answer 14 (response from Brian Sheron, NRC): Immediate emphasis has been placed on those issues raised in NRC's "The Near-term Task Force Review of Insights from the Fukushima Dai-ichi Accident" and the March 2012 NRC orders to operating nuclear power plants. However, additional research into advanced light water reactor fuel designs is being conducted as part of NRC's Long-Term Research Program. These studies are examining the licensing, acceptance, testing, and surveillance criteria to be used for advanced fuel materials including cladding that are more resistant to damage under accident conditions. The work is being conducted in concert with the U.S. Department of Energy, which is sponsoring the identification and development of these new designs. Proposed testing is extensive and will ultimately involve Lead Test Rod/Lead Test Assembly irradiations prior to commercial deployment.

TH26 *- Conservatism in Decision Making (Recent Operating Experience)*

Session Chair: John Thorp, NRR

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The questions below were not answered during the above session:

Question 1: Besides special inspections, what additional actions is the NRC considering to examine operational knowledge at [nuclear plants]?

Answer 1: In December 2011, NRC staff updated Inspection Procedure 711111.11, "Licensed Operator Requalification and Licensed Operator Performance," to include 4 hours per quarter of control room observation by either the resident inspector or a licensing examiner. These observations are meant to give inspectors a feel for 'operational knowledge' of the licensee's control room crews.

Question 2: Standards Drift' can be very subtle and occur over a long period of time. Have you developed any specific actions that will help you with early detection and intervention for eroding standards over the long term?

Answer 2: You are correct that "standards drift" is subtle and occurs over time. There are a number of actions that are taken to identify and correct drift. Oconee's Performance Improvement process serves as a first line of defense to identify early indication of drift. In addition to the performance improvement process, audit functions and assessment activities serve to recognize and correct drift.

In 2010-2011, Millstone Station implemented a program called Leadership High-Intensity Training (LHIT) as part of its Back to Basics efforts. The focus of LHIT is to reinforce standards and expectations for supervisors based on existing guidance. Initial LHIT training involved senior plant management delivering a 4-day session of scenarios and situational training to field supervisors. The training culminated with an oral board evaluation of each supervisor and a follow-up evaluation in the field. Millstone intends to make LHIT a "living" program and hold regular follow-up sessions to maintain supervisors' familiarity with Dominion's standards and expectations.

***TH27** – The NRC's Safety Culture Policy Statement and Agency Initiatives*

Session Chair: Diane Sieracki, OE

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The questions below were not answered during the above session:

Question 1: With the NSC policy in place 6+ months, how long until cross-cutting guidance can be aligned to the policy statement?

Answer 1: To date, staff has identified the ROP guidance and inspection documents that may need to be updated to reflect the Commission's safety culture policy statement. Many of these documents will also need to be updated after the common language initiative is complete. Therefore, to maximize efficiency, the NRC will undertake updating the appropriate documents after the common language effort is complete.

Question 2: Why only enhance safety culture within the ROP? Why not formally regulate safety culture? (It seems unlikely that the new safety culture process would have prevented the Davis-Besse reactor vessel head issue.)

Answer 2: In 2006 the ROP was modified at the direction of the Commission to enhance the ROP's treatment of cross-cutting issues to more fully address safety culture. The current inspection and performance assessment programs are effective at providing early indications of declining licensee performance. The ROP's treatment of cross-cutting issues has provided performance insights in cross-cutting areas that otherwise would not have been considered during the performance assessment process. The ROP also provides for a graded approach to assessing safety culture when performance warrants based on progressive decline across the action matrix.

Question 3: Do you or have you considered the use of focus groups to elicit safety culture issues among professional staff as well as nuclear plant workers?

Answer 3: The NRC uses a variety of techniques to evaluate a licensee's safety culture. These techniques include the use of focus groups, even for professional staff.

Question 4: Is the safety culture assessor qualification card publically available? Who is being qualified?

Answer 4: The safety culture assessor qualification card is publically available on the NRC's public website. You can find it at the following location if you scroll down the page to IMC 1245 App C12: http://www.nrc.gov/reading-rm/doc-collections/insp-manual/manual-chapter/index.html.

The NRC qualification process was developed for NRC inspectors. Staff pursuing safety culture assessor certification do not need to be previously qualified as an inspector since safety culture assessors work as part of a team on inspections. However, all staff pursuing safety culture certification is expected to:

1) Understand the legal basis for and the regulatory processes used to achieve the NRC's regulatory objectives

2) Master the techniques and skills needed to collect, analyze, and integrate information using a safety culture focus to develop a supportable regulatory conclusion

3) Demonstrate the personal and interpersonal skills needed to carry out assigned regulatory activities, either individually or as part of a team

Question 5: A) How are the 13 safety culture components of the ROP correlated to the 9 safety culture traits defined in the safety culture policy statement? B) If they are not correlated at the moment, are you planning to revise the ROP to reflect the content and character of the safety culture policy statement?

Answer 5: Currently, the safety culture components in the ROP are not correlated to the safety culture traits in the safety culture policy statement. To date staff has identified the ROP guidance and inspection documents that may need to be updated to reflect the Commission's safety culture policy statement. Many of these documents will also need to be updated after the common language initiative is complete. Therefore, to maximize efficiency, the NRC will undertake updating the appropriate documents after the common language effort is complete.

Question 6: NEI 09-07 has been implemented industry-wide. My experience and peer comments have indicated the process is not meeting its intended results. What changes are planned by NEI to address process weaknesses?

Answer 6: Unfortunately, the questioner did not indicate the basis for the judgment that the NEI 09-07 process is not meeting its intended results. The questioner was invited to communicate with the NEI speaker privately after the session to discuss his or her concerns more thoroughly. As of April 6, 2012, this contact has not occurred.

For perspective, it is important to keep in mind that the safety culture initiative was implemented October 1, 2011, as committed by industry. According to the NEI 09-07 process, sites have up to six months to hold the first meeting of the senior leadership team. Thus by the time of the RIC, some sites may have held only their first senior leadership team meeting. Given this, it appears premature to judge that the process is not meeting its intended results and premature to talk about changing the process.

INPO is evaluating implementation of the NEI 09-07 process. NEI will monitor industry performance for signs of culture-related incidents, as a measure of the effectiveness of the NEI 09-07 process. Whether and how to modify the NEI 09-07 process will depend on what that data reveals.

Question 7: How much does the Nuclear Safety Culture Assessment aligned [sic] with the context of Safety Culture Policy Statement? For example, are 9 s[sic] traits in the policy statement incorporated?

Answer 7: The NSCA was written to align with the INPO Principles for a Strong Nuclear Safety Culture, which pre-dates the NRC's Safety Culture Policy Statement. Nevertheless, the NRC's SCPS and the elements of the NSCA have a lot in common. When the Common Language is completed (hopefully, before the end of 2012), the alignment of the NSCA and SCPS will be clear. At that time, industry may modify the wording of the NSCA to ensure alignment with the Common Language.

Question 8: Can you discuss further the faint signals and how they are identified?

Answer 8: So-called faint signals are the anecdotes and other precursors that appear before one can quantify a trend with numerical indicators. They may take the form of stories or episodes about choices individuals made that compromised the priority given to nuclear safety. In the "faintest" of those stories, the tie to safety culture or to compromising on the priority given to nuclear safety may be difficult to discern. Yet when probed deeply enough, or when a number of these "faint signal" stories are tied together, the sign of threats to the priority given to nuclear safety may become visible to those looking for a specific tie to safety culture. In one apocryphal story, an individual acquiesced to what he perceived to be pressure to meet schedule, despite lingering questions in his mind about the safety decision that was being discussed. When this critical moment was later examined more deeply, the decision-maker realized that there remained a residual reluctance to question the decisions of a superior, despite significant efforts made to demonstrate that senior management truly was "walking the talk" about nuclear safety being the organization's top priority. This insight led to redoubled efforts to make the atmosphere safe and fault-free for raising questions about the priority given to nuclear safety at all times.

Question 9: Will the results of the Nuclear Safety Culture Assessment pilot evaluation be made public through the outreach activities of NET [sic]?

Answer 9: The results of the NSCA belong to the host site/company. Decisions about making the NSCA results public belong to the host site/company alone. One NSCA report has been put on an NRC docket, and is available through ADAMS. That is a 2010 NSCA of the Prairie Island Nuclear Generating Station, available under ADAMS Accession Number ML102460761.

Question 10: What efforts, if any, are underway to identify future industry-wide challenges to maintaining strong safety cultures in the nuclear industry?

Answer 10: INPO is evaluating implementation of the NEI 09-07 process and will draw insights from its evaluations. NEI will monitor industry performance for signs of culture-related incidents, as a measure of the effectiveness of the NEI 09-07 process. As data becomes available, NEI and INPO will collaborate to identify industry-wide challenges to maintaining strong safety cultures in the nuclear industry.

Question 11: Will NSCA process use the INPO SC assessment process or some subset of it?

Answer 11: The NSCA process uses the INPO Principles of a Strong Nuclear Safety Culture as the basis for assessing safety culture. The process for completing the NSCA has some high level similarities to the INPO safety culture evaluation process, but varies significantly in the details.

Question 12: What are some of the specific examples of the "common language" issues? What concepts/terms could cause problems?

Answer 12: The issues are as simple and profound as the number and wording of elements used in defining safety culture. The NRC's Inspection Manual speaks of 13 components and 35 aspects of safety culture; INPO speaks of eight principles and 50-plus attributes of safety culture. Thus, when site staff discusses safety culture with the NRC, they must do so in terms of the components and aspects presented in NRC's documents. When site staff discusses safety culture with INPO, they must do so in terms of principles and attributes. This "double duty" adds to difficulty in communicating about safety culture. The common language project seeks to reconcile these differences by developing a structure of the elements comprising safety culture and definitions for those elements that NRC and industry (INPO and NEI) can agree will work for all parties.

TH28 *- Thermal-Hydraulic and Severe Accident Research*

Session Chair: Michael Scott, RES

Session Coordinator: Antony Calvo, RES
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Questions submitted during the above session were answered during the session's Q/A period.

TECHNICAL SESSIONS
Thursday, March 15, 2012, 10:30 a.m. 12:00 p.m.

TH29 *- Allegations, Alternative Dispute Resolution, and Enforcement: Statistics and Initiatives*

Session Chair: Roy Zimmerman, OE

Session Coordinator: Lauren Casey, OE
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The questions below were not answered during the above session:

Question 1: ”Deemed” exports is a significant topic these days. Will the proposed examples of import/export of regulated materials include “deemed” export examples? Can we expect more guidance on “deemed” exports?

Answer 1: The NRC does not license “deemed exports”. Both the Departments of Energy and Commerce have requirements regarding “deemed exports”. Under the "deemed export rule" in the Department of Commerce’s Export Administration Regulations (EAR), a transfer of technology or computer source code to a foreign national is "deemed" to be an export to the home country of the foreign national. Similarly, the Department of Energy has authorization requirements for transfers to foreign nationals of technology related to the production of special nuclear material under its regulations in 10 CFR Part 810.

Because deemed exports do not fall under NRC’s export licensing authority, questions related to Commerce’s EAR can be directed to: 202-482-4811. Questions related to Energy’s 10 CFR Part 810 can be directed to: Jo Anna Sellen Bredenkamp at Joanna.Sellen@nnsa.doe.gov or sellenjm@ornl.gov.

Question 2: Does OE approve issuance of CEL’s by the regions? If not, does OE audit the issuance of CEL’s and decisions of not issuing CEL’s?

Answer 2: The recently issued guidance in Allegation Guidance Memorandum 2012-001, “NRC Chilling Effect Letters” (Agencywide Documents Access & Management System Accession No. ML 12025A055) directs that the OE Director or his or her designee shall be on the CEL’s concurrence, while the signature authority remains with the applicable Regional Administrator, Office Director, or his or her designee.

Question 3: Some allegers raise very high numbers of allegations that are not substantiated. Are there any ways that licensees can limit resource expenditures on unsubstantiated allegations?

Answer 3: There are a number of underlying reasons concerned individuals might raise concerns outside of their organizations, including but not limited to, perceptions associated with management behaviors, self-interests, fear of retaliation, and weaknesses in internal processes. Organizations that establish and maintain effective processes to mitigate or eliminate such barriers to a safety conscious work environment (SCWE), including countering perceptions with truthful and effective communications, typically limit their exposure to large numbers of allegations. While it is acknowledged that considerable effort can be expended in evaluating concerns that are ultimately determined to be unsubstantiated, encouraging the free flow of information by being responsive to all concerns, valid or otherwise, and identifying insights into an organization's SCWE resulting from the concerns raised, are some of the benefits realized by maintaining a low reporting threshold.

Regarding the statement that some allegers raise very high numbers of unsubstantiated allegations, it is important to understand that under current policy every new concern must be evaluated on its own merit. The validity of previously raised concerns is not predictive of the validity of a new and different concern raised by the same individual. Finally, it would be contrary to the underlying principals of the NRC's safety and security mission, as well as a licensee's commitment to a strong safety culture, to limit the number of concerns an individual can raise, or to protect only those who raise valid concerns.

Question 4: Suggest putting ADAMS accession numbers for all documents mentioned on slides.

Answer 4: Agree. Below please find the Agencywide Documents Access & Management System (ADAMS) Accession numbers for those documents referenced on the slides during this panel:

- Allegation Guidance Memorandum 2012-001, "NRC Chilling Effect Letters" ML 12025A055

- SECY-11-0155, "Proposed Changes to the Enforcement Policy Associated with Construction Activities" ML11293A028

Question 5: Can a CEL also be sent to a non-licensee involved with nuclear activities? For example, a vendor, a supplier of a safety-related component?

Answer 5: The NRC's May 4, 1996 Safety Conscious Work Environment (SCWE) Policy Statement, "Freedom of Employees in the Nuclear Industry to Raise Safety Concerns Without Fear of Retaliation," (61 FR 24336 or http://www.nrc.gov/about-nrc/regulatory/allegations/scwe-frn-5-14-96.pdf), and August 25, 2005 Regulatory Issue Summary 2005-18, "Guidance for Establishing and Maintaining a Safety Conscious Work Environment," (http://www.nrc.gov/reading-rm/doc-collections/gen-comm/reg-issues/2005/ri200518.pdf) establish the NRC's expectations that NRC licensees, holders of certificates of compliance, applicants, and their contractors, subcontractors, and other employers subject to NRC authority, establish and maintain an environment for raising concerns without fear of retaliation. Therefore, if the NRC becomes concerned about the SCWE at any such employer, it can issue that employer a CEL, and has in the past issued such a CEL in appropriate circumstances.

Question 6: Is OGC involved in the mediation process? Must OGC approve the mediated settlement?

Answer 6: Yes. In Early ADR, OGC reviews the settlement agreement to ensure that no restrictive agreements in violation of the applicable NRC Employee Protection Rules (e.g. 10 CFR 50.7(f)) are contained in the settlement agreement. In Post-Investigation ADR, an OGC representative is involved throughout the process and is a member of the mediation team who also reviews the ultimate confirmatory order.

Question 7: Does Cornell usually provide the same individual as the mediator? That is, is the mediator familiar with NRC environment?

Answer 7: Cornell has a roster of mediators who are located in different geographical areas of the country. Depending on the venue of the mediation, Cornell typically provides the name of three mediators from the general vicinity of the mediation venue. The parties to the mediation mutually select their mediator from that list. Mediators on Cornell's roster are generally knowledgeable about the NRC.

Question 8: Why was there a spike in 2009 for post investigation ADRs?

Answer 8: The spike in 2009 is attributable to a greater than usual number of individual and materials licensee cases.

Question 9: Compliance is mandatory. Why should a violator get "credit" i.e., a reduced penalty for belated compliance (corrective actions)?

Answer 9: Compliance with NRC requirements provides reasonable assurance to the NRC and the public that safety and security are being maintained. Whenever the NRC becomes aware of a noncompliance, the first priority is to ensure that public health, safety and security is not compromised. The NRC, then, determines the appropriate enforcement action warranted, if any, in instances where noncompliance is present. In determining the appropriate enforcement action, the NRC recognizes the importance of the licensee correcting the noncompliance and of encouraging prompt and comprehensive corrective actions, in addition to other factors. Granting corrective action credit encourages the completion of actions that (1) restore safety, security and compliance and (2) that address actions to prevent recurrence of the violation at issue and occurrence of violations with similar root causes (which is not mandatory in all cases). In recognition of the importance given to corrective actions, an NRC judgment that the licensee's corrective action has not been prompt and comprehensive will, typically, result in the issuance of at least a base civil penalty.

Question 10: What about ADR at the ASLBP? 10 CFR § 2.338 promotes ADR at the ASLBP but boards cannot mandate it. Why doesn't NRC staff use this ADR tool?

Answer 10: The adjudicatory procedures encourage the use of ADR to resolve issues in litigation. The NRC staff, as a party in litigation, has used the procedures in 10 C.F.R. § 2.338 in settlement negotiations. These procedures provide for the appointment of a Settlement Judge by the Chief Administrative Judge of the ASLBP upon a request by the presiding officer of an adjudication. It is more common, however, for the NRC staff to participate in settlement negotiations of adjudicatory proceedings without the assistance of a Settlement Judge. Regardless of the form of negotiations, all settlements of adjudicatory proceedings must be approved by the presiding officer.

TH30 *- Construction Inspection Program—Transitioning to Execution*

Session Chair: Laura Dudes, NRO

Session Coordinator: Victor Hall, NRO
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The questions below were not answered during the above session:

Question 1: A large number of Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) items occur during the initial test program phase of the project. Has the impact of ITAAC compliance on executing a Regulatory Guide 1.68 startup test program with respect to process schedule and resources been considered? Do you have any thoughts or recommendations?

Answer 1: Yes, a large number of ITAAC are performed during the initial test program phase of the project, and the ITAAC requirements are being integrated directly into the Regulatory Guide 1.68 "Initial Test Programs for Water-Cooled Nuclear Power Plants". ITAAC are integrated into the comprehensive testing requirements and performed as part of overall system preoperational testing procedures. Note that most of the ITAAC in this phase are tests that are already performed under any Regulatory Guide 1.68 test program and are identified in the Design Control Document (DCD) Tier 2 Chapter 14, just with a new regulatory significance. Therefore, schedule and resource considerations are factored directly into the planning process, rather than a separate effort in addition to the test program.

Question 2: How do you assure that your vendors are adequately qualified?

Answer 2: All licensees, applicants, and their safety-related vendors are required to establish and implement a Quality Assurance program that meets the requirements of Appendix B to 10 CFR Part 50 "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants". A comprehensive system of planned and periodic audits is included as part these requirements. A successful audit is required for licensees and applicants to maintain a vendor on its approved supplier list (ASL).

Question 3: Does the applicant perform third-party oversight of its contractors and vendors to ensure the vendors are qualified and can meet the purchase order requirements and have an effective Part 50 Appendix B Quality Assurance program?

Answer 3: As mentioned in the previous question, a successful audit is required for vendor qualification. The Nuclear Procurement Issues Committee (NUPIC) is a third party audit organizations that provides oversight.

Question 4: What formal processes is Shaw using to share lessons learned on an ongoing basis between the concurrent Vogtle and V.C. Summer projects (e.g. are there periodic meetings between project leadership teams)?

Answer 4: Shaw Nuclear uses several formal processes along with an aligned organizational structure to share construction experiences from the ongoing nuclear projects. The Lessons Learned process database is integrated into Readiness Review process planning meetings supported by both the Vogtle and V.C. Summer organizations. Performance Improvement and Lessons Learned Managers are assigned at each project to provide input and also lead in capturing ongoing construction experiences. One current practical example of sharing and gaining construction experience is the First Nuclear Concrete Mockup where both sites are participating.

Question 5: What license amendment and departure workload will occur following Combined Operating License (COL) receipt and how have you prepared/staffed for it? How do you ensure construction does not get ahead of licensing basis?

Answer 5: To date there are approximately 20 License Amendments and several hundred departures to process. Personnel involved in supporting the NRC COL review and response to Requests for Additional Information (RAIs) are now supporting this effort. SCE&G and Consortium personnel are utilizing a master list of changes with an associated construction need date to manage the work.

Question 6: When do you think V.C. Summer will receive its COL?

Answer 6: On March 30, 2012, the NRC concluded its mandatory hearing on the South Carolina Electric & Gas (SCE&G) and Santee Cooper application and issued two COLs for the V.C. Summer site in South Carolina.

Question 7: What is SCE&G doing to ensure that construction is done per the license? Is training being used? How is alignment being achieved with the constructor?

Answer 7: By contract, SCE&G acceptance is required for any changes that affect the licensing basis. Procedures are in place between the Consortium and SCE&G to identify and process changes. Training has been conducted and ongoing oversight is provided by SCE&G personnel.

Question 8: How much money for Vogtle Units 3 and 4 was obtained from U.S. taxpayers (Federal Subsidies)?

Answer 8: The Department of Energy (DOE) has offered conditional loan guarantees to encourage the commercial use of new or significantly improved energy technologies. The federally-backed guarantees are intended to help would-be builders to raise private finance at no cost to the taxpayer.

Question 9: Regarding the "work packages" you mentioned during your presentation, what is your estimate for the total number of work packages to build Vogtle Units 3 and 4? How has this estimate changed over time?

Answer 9: The current estimate is approximately 10,000 work packages per unit.

Question 10: For the total plant, what do you believe is the percentage of design construction at the time the RCOLA was issued? How do you see this changing for future plants that will follow Vogtle? How does this compare to the percentage of design complete for the Design Certification?

Answer 10: The RCOLA was issued within 2 months after the Design Certification Amendment so the status of the design completion was essentially the same for both milestones. At that time, the standard plant nuclear island design was approximately 90 to 95% complete. For future plants, it is expected that the standard plant design will be greater than 95% complete.

Question 11: Can you elaborate on Preliminary Amendment Requests (PARs) related to the Vogtle Combined Operating License?

Answer 11: A licensee can request a PAR to mitigate the risk to scheduled construction activities when the License Amendment Request (LAR) may not be approved prior to the scheduled construction activity.

Question 12: Can you describe the division of responsibilities between NRC's Office of New Reactors (NRO) and Office of Nuclear Reactor Regulation (NRR) with respect to developing and overseeing construction inspection programs for new reactors being licensed and built under the Part 50 process (e.g. Small Modular Reactors (SMRs), Bellefonte, and Watts Bar 2)?

Answer 12: NRO is responsible for developing the construction inspection programs for new reactors and SMRs. NRR is responsible for developing the construction inspection programs for Bellefonte and Watts Bar 2.)

Question 13: Do the construction related parts of 10 CFR Part 50 apply to Part 52 licensees (e.g. 50.55(e) significant construction deficiency reporting)?

Answer 13: Yes.

Question 14: How do ITAACs relate to non-safety-related structures within a new nuclear facility?

Answer 14: ITAAC exist if they're necessary and sufficient to ensure the plant is constructed and will operate in accordance with its design and applicable regulations. ITAAC are commensurate to the safety and risk significance of the system. There are non-safety related structures, systems, and components that have ITAAC. An example of this would be the AP1000 startup feed-water system, which is expected to be available as a non-safety-related system to provide startup feed-water flow from the condensate storage tank (CST) to the steam generator system (SGS) for heat removal from the Reactor Coolant System (RCS).

Question 15: What criteria are used to select which specified ITAACs will be verified by the NRC staff? Which document describes the selection criteria?

Answer 15: The NRC will inspect a sample of the ITAAC. This sample is commonly referred at the targeted ITAAC. The targeted ITAAC were chosen for inspection based upon a prioritization process presented in Commission Paper SECY-07-0047 "Staff Approach to Verifying the Closure of Inspections, Tests, Analyses, and Acceptance Criteria through a Sample-Based Inspection Program".

***TH31** - Current Reactor Oversight Process Issues of Interest to the Public, Staff, and Industry*

Session Chair: John Lubinski, NRR

Session Coordinator: Michael Balazik, NRR
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The questions below <u>were/were not</u> answered during the above session:

Question 1: When security is added back into the matrix will GTG issues going back for a 4 quarter period be reflected as only security issues documented after the 7/12 date? The webpage matrix shows the current and last 3 quarters-will security only show those after the reintegration takes place?

Answer 1: The webpage will reflect security findings as PIs for all four quarters. As I mentioned in my presentations, inspection findings and performance indicators on the NRC's web pages may appear to be new. But in most cases these inputs to the action matrix existed before the webpages were modified, and the NRC has been applying appropriate levels of response.

Question 2: Reintegration of security in the ROP is a bit of a misnomer since security is treated deterministically within the SDP process. Has the staff considered only applying traditional enforcement for security violations/findings?

Answer 2: The reintegration pertains to the assessment program. The security cornerstone has always remained part of the ROP framework, even though a separate assessment program has applied to security inspection findings and performance indicators since 2005. However, the security cornerstone is not unique in its application of deterministic significance determination tools. The two radiation safety cornerstones and the emergency preparedness cornerstone similarly apply deterministic significance determination tools. Traditional enforcement is used for all ROP cornerstones but is reserved for violations that (1) impede the regulatory process (i.e., 10 CFR 50.9 violations and reporting violations), (2) are willful, and (3) have actual consequences.

Question 3: Why won't the NRC show the color of greater than green findings or PI's? The impact to the action Matrix column will often reveal the color anyways, won't it?

Answer 3: The staff's proposed to reintegrate the safety and security assessment programs; this proposal did not extend to the Commission's current policy regarding the protection of sensitive, non-safeguards information. As such, the Commission approved changes to the assessment programs only. The Commission's policy regarding information disclosure remains in effect, unchanged. Since cover letters for supplemental inspection reports are publicly available and currently reveal the specific supplemental inspection that was performed (even within the security cornerstone), it is possible to surmise the significance of inputs that generated the regulatory response. That potential to surmise the actual color of a greater-than-green finding will continue to exist upon reintegration.

Question 4: Much nuclear power plant information of value to the public is considered proprietary to licensees. What are the impacts on a full public dialog about nuclear safety?

Answer 4: This question is difficult to answer without knowing, more specifically, the proprietary information of interest. The impact of divulging trade secrets is that a licensee would potentially loose a competitive edge over its cohort in the industry that could affect their standing in the marketplace. That said, provisions for protecting security information are different for those governing proprietary information (See Dave Lochbaum's and Joel Munday's answer for additional insights related to the handling of security and proprietary information). The sensitive, security information that is withheld from public disclosure is not, per se, proprietary. These are two distinctly different classifications of information that require special treatment in accordance with NRC policy and/or regulations.

Question 5: What shade of Green will TVA get for working on the wrong reactor?

Answer 5: The presenter is not cognizant of inspection findings that are being developed in the regions. That said, if the inspection finding is not documented in an inspection report or final significance determination letter, then it is pre-decisional information upon which the presenter (or any other member of the NRC staff) would not be able to comment.

Question 6: It's been said that NRC conducts only 5-10% of NRC mandated inspections itself, the rest are done by licensees. Is that enough?

Answer 6: In accordance with the tenets of the Reactor Oversight Process, the NRC conducts inspections on a sampling basis to assess a licensee's safety performance, compliance with regulatory requirements, and conformance to the facility operating license. A fundamental goal of the NRC's reactor inspection and assessment process is to establish confidence that each licensee is detecting and correcting problems in a manner that limits the risk to members of the public. As such, licensees conduct inspections (like bare metal visual inspections to evaluate the material condition of the reactor vessel head) to satisfy regulatory requirements and the NRC assesses the licensee's performance to ensure that the licensee is satisfying the requirements. Licensees also evaluate systems, structures, components, and programs to ensure they are effective in achieving their safety functions. To this end licensees may perform inspections, calculations, analyses, tests, and other actions that support facility operation. Is this enough? The NRC staff performs an evaluation of the ROP and associated inspection program every two years to ensure that inspection resources are allocated to those areas where optimum safety benefit can be achieved. Resources are routinely adjusted based on lessons learned to achieve this "optimization".

Question 7 (addressed to Dave Lochbaum, UCS): Much nuclear power plant information of value to the public is considered proprietary to licensees. What are the impacts on a full public dialog about nuclear safety?

Answer 7: There can and should be a balance between information available to the public and information withheld from the public arena for security reasons. The winter Olympics held in early 2002 in Utah showed how it should be done. There was considerable media coverage of the security measures put in place to protect competitors and attendees at the games, so soon after the 9/11 tragedy. Bomb-sniffing dogs and metal detectors and such were publicly discussed so people going to the games would not be unduly afraid. But the specifics of the security measures was appropriately withheld so that people wishing to cause harm would not be aided in planning their attacks. Similarly, the NRC can balance the public's need to feel adequately protected from attacks on the nuclear facilities in their communities with the need to withhold information that might add our enemies. Restoring information to the public ROP is appropriate because it will not reveal current security vulnerabilities that could aid our enemies.

Question 8 (addressed to Dave Lochbaum, UCS): How does the ROP provide reasonable assurance that a plant is in compliance with their current licensing basis and 10 CFR 50?

Answer 8: NRC inspection procedure 71111.21 (the component design bases inspection) seeks to do this. But it falls short, in my view, because the NRC does not ask and licensees do not answer the fundamental question of why the licensee's testing, inspection, and configuration management regimes all failed to find the design basis problems identified by the NRC inspectors. The NRC's CDBI is a very, very, very limited audit - more like a drive-by peek - of design basis. It is not independently verifying that plants comply with CLB. Instead, it seeks to assess whether licensees are maintaining their facilities within their CLB. Each CDBI finding reflects a breakdown in the licensee's CLB efforts. But the CBDI does not assure that those weaknesses are corrected. Thus, the CDBI does not correct the systematic/programmatic problems resulting in the NRC CDBI findings. Close but no cigar.

Question 9 (addressed to Michael Llewellyn, INPO): Much nuclear power plant information of value to the public is considered proprietary to licensees. What are the impacts on a full public dialog about nuclear safety?

Answer 9: There is a significant amount of information about nuclear plant operation and nuclear safety available to the public via the U.S. Nuclear Regulatory Commission (NRC). Much of the information gained through NRC inspections and other activities is available to the public via the NRC Website. The NRC also enables additional public dialogue about nuclear safety via a variety of public meetings conducted at each U.S. nuclear plant. It is vital to the success of INPO's mission that the Institute hold private certain interactions with its members. Without confidence in the privacy of INPO interactions, the openness and candor between utility and INPO personnel involved would be significantly reduced. This would inhibit INPO's ability to obtain detailed information from utilities. The effectiveness and value of INPO's evaluations could be impaired, with a long-term consequence of reducing the margin of safety in the nation's nuclear plants.

Question 10 (addressed to Michael Llewellyn, INPO): Has INPO looked into the proposals for revised crew expectations for innovative reactors and control room designs for small modular reactors?

Answer 10: INPO personnel are active with small modular reactor activities, including control room operator staffing and licensing, through the NEI-sponsored small-modular reactor task force.

Question 11 (addressed to Michael Llewellyn, INPO): Will you consider additional risk-relevant scenarios for operational crew performance evaluations?

Answer 11: A factor in our selection of simulator exercises for the crew performance evaluation process considers operator actions identified as important by PRA analysis. This information for each station is reviewed by INPO and used to inform the selection of exercises used for crew evaluation.

Question 12 (addressed to Michael Llewellyn, INPO): Has INPO considered ways of making operational event information public without releasing all the details in a SOER?

Answer 12: This information in summary form would be useful. A significant amount of event information is available from the NRC. The value INPO brings is in providing detailed information and analysis results to plant operators who are in a position to use this information to potentially prevent similar events, and, overall, to improve the margin of nuclear safety. Much of this detailed information is proprietary and, therefore, event details and analysis results available to our members through documents such as Significant Operating Experience Reports (SOERs) cannot be released publicly.

Question 13: Much nuclear power plant information of value to the public is considered proprietary to licensees. What are the impacts on a full public dialog about nuclear safety?

Answer 13: When it comes to discussing NRC practices, processes and licensing decision actions, the agency has been an advocate for openness in an effort to foster transparency, participation, and collaboration in our regulatory activities. However, we are not at liberty to provide or discuss with the public nuclear safety information that a licensee has categorized as proprietary. The agency does not control or regulate information that is deemed proprietary. The impact of discussing such information could place the agency in a position that could lead to a compromise of technological or business information. When the NRC reviews information that is considered proprietary by a licensee, the licensee must submit an application for withholding pursuant to 10 CFR 2.390, "Public inspections, exemptions, requests for withholding," along with a required affidavit to support withholding the information from public disclosure. Often, non-proprietary versions of documents are made available to the public to provide information on regulatory decisions about nuclear safety.

Question 14: How are the weaknesses that are being seen in the training of operations being communicated to both industry and NRC inspectors? What actions have the plants been taking to address them?

Answer 14: A number of recent industry events have occurred where operator actions were identified as a contributing factor. The agency has reviewed its processes and has revised the inspection procedure for operator requalification (IP 71111, Attachment 11) to better address aspects of training and to identify operator performance issues at a lower threshold. The procedure was effective in January 2012 and is being used by resident inspectors and operator licensing examiners to conduct more in-depth reviews of licensee programs for the training of operators and to more closely monitor the conduct of operations. The agency also recently conducted forums at the RIC and plant manager and utility user group meetings to address these issues. The topic was also discussed at public meetings with the Institute of Nuclear Power Operations (INPO) and the Nuclear Energy Institute Operator Licensing Focus Group. In addition, the industry, through INPO, has undertaken its own initiatives to capture and address these concerns. They have issued documents to the utilities to communicate recent observations, required licensees to review their programs, and have recommended enhancements to better fortify and evaluate operator training programs.

Question 15: Nine transformer failures seem to be an aging problem. Is the NRC considered aging issues in Operator Requalification on older plants?

Answer 15: The aging of plant components and events caused as a result of aging issues is real and is a consideration in the development of operator training scenarios, but licensees are not explicitly required to include all aspects of operational events, such as aging, in their training programs. The agency requires that licensees develop a training program that ensures operators can operate safely and appropriately respond to events. Among many other topics, licensed operators are trained annually on loss of electrical power as required by 10 CFR 55.59, "Requalification". While inclusion of aging events, natural phenomenon, or beyond design basis events is not specifically addressed in the requirements for an approved systems approach to training program, industry training programs are required to consider relevant operating experience in training development. Given recent events and operator trends, more consideration of these and other topics is being evaluated.

Question 16: You skipped over “conservative decision-making” due to lack of time. Can you say more about how you define that and what’s the appropriate degree of conservatism?

Answer 16: Conservative decision making is best characterized by operating crews making decisions based on approved procedures, correct interpretation of the procedure’s intent, the ability to question the appropriateness of the intended actions, being aware of the importance of certain operating limits and the ability to maintain the design configuration of the plant. In the context of responding to an event, conservative decision making would dictate that if an operating crew is faced with a situation where the plant does not respond as expected, the evolution would be stopped and the unit placed in a safe and stable condition until conditions are fully understood and an approved path forward is developed. In some instances, this could include shutting down the plant. If operations are conducted with these attributes, then conservative decision making principles likely have been met. Please refer to the materials associated with RIC session TH26, “Conservatism in Decision Making (Recent Operating Experience)” for more information on this topic.

Question 17 (addressed to Christopher Earls, NEI, and Ralph Andersen, NEI): Much nuclear power plant information of value to the public is considered proprietary to licensees. What are the impacts on a full public dialog about nuclear safety?

Answer 17: The NRC requires licensees through 10 CFR Part 73.21 to safeguard certain security-related information to help ensure that terrorists and other potential adversaries to the site security forces do not have specific details about the security equipment and plans which could be used to plan and execute attacks on the plant. Given these restrictions, it is very difficult to have a meaningful public discussion about specific security-related issues. In addition, there is a concern that any specific discussion of identified weaknesses or vulnerabilities could be exploited despite the fact that the site is required to put compensatory measures in place until the issue is fully resolved. This approach to safeguarding sensitive security-related information is not unique to the commercial nuclear power industry and is common practice at Federally mandated secured facilities.

***TH32** - License Renewal—Perspectives on Current and Subsequent License Renewal, Part 2 of 2 (Double Session with TH23)*

Session Chair: Michael Case, RES

Session Coordinator: Greg Oberson, RES
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Questions submitted during the above session were answered during the session’s Q/A period.

***TH33** – Radiation Protection*

Session Chair: Stephanie Bush-Goddard, RES

Session Coordinator: Gladys Figueroa, RES
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Question 1 (addressed to Don Cool, NRC): In your survey of the audience you asked the audience about whether they would shelter in place and got a majority, yes. Shouldn't you have asked whether people would decide on a 2 rem dose in the 1st year depending on whether they had children? What would you do if you had a one year old daughter and a prospect of 2 rem dose?

Answer 1: This is a good question. Radiological guidelines for public protection are very conservative. Limits like 100 mrem/y for the public are protective for situations that are planned based on the ALARA (as low as reasonably achievable) principle. They are far below limits set for radiation workers, 5000 mrem/y, which is still a level considered quite safe. In an emergency situation, higher levels than normal can be tolerated as still safe, especially when the alternative is that towns and livelihoods are abandoned. Some balance between concern for children and loss of home and possibly job, may be achieved by using the shielding afforded by the homes, schools, and community buildings. During the second year, cesiums deliver most of the dose through groundshine. We recognize that it is the adult that is most sensitive to cesium intake. If groundshine is the only factor, children may be more sensitive, but we're still learning about low doses and distinctions between adults and children. There is some evidence that low doses actually stimulate the immune system. There are geographical areas of high background radiation with no apparent increase in risk. Will the increase in medical use of radiation result in increased mortality from cancer? Where people reside in contaminated areas of Japan, parents can inform and make good decisions for themselves and their children. Children, informed by adults, avoided hotspots in parts of Belarus when returning from school. What would I personally do? At the dose level of 2 rem, I would prefer to stay and help the community recover, adopting practices that would minimize exposure to my family, but I would not be any more concerned about the radiation to my daughter than I would be if a doctor suggested that she needed a CAT scan.

In a recent PBS documentary on Fukushima, a Japanese government employee was asked what was the increased risk of cancer from living in a 20 µSv (2 rem) zone. He replied 0.2. When asked what was the normal risk for Japanese people, he replied 30. So another perspective is an increased risk from 30.0 to 30.2, maybe slightly more for children. Would you give up your home and livelihood for that margin of safety? Everyone has a choice, if permitted.

Question 2 (addressed to Steven Garry, NRC) : EPA has commissioned and recently completed studies related to the radiological effects of tritium. The study has shown that the biological effective half life is higher than previously thought. This will change the dose coefficient associated with tritium. How will this potentially effect NRC's staff recommendation to the Commission regarding Regulatory Framework?

Answer 2: EPA is sponsoring Oak Ridge National Laboratory (ORNL)'s updating of the biokinetic models that describe radionuclide behavior in the human body, as part of a larger effort to update the International Commission of Radiation Protection (ICRP)'s biokinetic models for *all* elements. This ORNL work will be considered by ICRP and might appear in future ICRP publications, as prepared by ORNL or modified by ICRP et al. The EPA report on the new tritium biokinetic has not been published by EPA and is under consideration by ICRP.

Question 3 (addressed to Brian Littleton, EPA): Current methodology includes organ dose (FGR.13). Why is the organ dose proposed to be abandoned in the name of updating the science? Won't the EDE approach greatly increase allowable organ dose from certain radionuclides like radioiodine, actinides, and Sr-90?

Answer 3: This was answered during the conference. Response – The Agency considers EDE/effective dose as the most current dosimetry and we seek to revise our requirements to the most current dosimetry. The effective dose equivalent/effective dose takes account of the relative sensitivity of different organs and tissues to radiation, and by extension the relative cancer risk presented by radiation exposure to those organs and tissues. Adopting EDE would bring us in line with more current science than the critical organ method, which is largely out of use and does not account for the latest risk information. There are several issues that have not been determined yet in moving to an EDE dosimetry, such as what will be the numerical standard. The Agency is seeking comment to help us make a determination on these issues.

Question 4 (addressed to Brian Littleton, EPA): If kept, kept do you envision the radiological release limits to change, will they?

a. Maintain the same units of Cs/Gw(e)
b. Allow values to change (up or down) as updated data is utilized
c. Include additional radionuclides
d. Allow enforcement per facility

Answer 4: This was answered during the conference. The Agency at this time does not have a pre-conceived notion of how the radionuclide release limit may change. We are very open to receiving comments and recommendations on how to change this portion of the standard to be more effective.

Question 5 (addressed to Brian Littleton, EPA): There are many controversial questions that are being asked, what sort of timeline do you expect of rulemaking to take?

Answer 5: This was answered during the conference. We anticipate getting a draft of the ANPR to the White House's Office of Management and Budget soon with hopes of the ANPR being published sometime this Spring 2012. The timing for a specific regulatory proposal, should the Agency pursue one, would depend on a number of factors, including the extent of public comment on the ANPR.

Question 6 (addressed to Brian Littleton, EPA): How will EPA and NRC coordinate their respective RP regulation change to ensure regulatory stability and no inconsistencies for licensees?

Answer 6: The EPA and the NRC have held more frequent meetings to keep each agency appraised of potential changes as we move forward with revisions to 40 CFR 190, 40 CFR 192, and potential changes to the NRC's radiation protection regulations. Although different statutory drivers may necessitate minor differences, the hope is to minimize these differences, to the degree possible.

Question 7 (addressed to Brian Littleton, EPA): The NRC mission includes Protecting Public Health and the environment. Could you please comment on the continued need for part 40? How can it be justified in the face of limited Federal Government resources? Would it not be better to combine Part 40 and Part20, is that possible?

Answer 7: Regulations developed under the authorities provided to the EPA are found in Title 40 of the Code of Federal Regulations. Specifically, regulations which govern environmental protection of nuclear power operations are in 40 CFR Part 190. The Reorganization Plan No. 3 states that it is the EPA's responsibility to develop these standards using Atomic Energy Act authority, and that they will be implemented by the NRC. The Agency does not have the authority to combine environmental protection standards with implementation regulations.

Question 8 (addressed to Brian Littleton, EPA): If alpha limits are to be imposed for Uranium fuel cycle, why are there no equivalent limits for fossil fuel. A little calculation will show the combustion of coal results in much more than 0.5 millicuries of RA-226/GW electric in coal ash that is produced.

Answer 8: Thank you for your comment. The Agency will consider this issue.

Question 9 (addressed to Brian Littleton, EPA): DOE produced a report (SANDIA Report) which provided a sort of technical basis for the original 40CFR190 and noted some updates that could be applied but did not apply these updates themselves, has EPA examined the impact of the updated information to the release limits and if so was the trend to make the release limit more restrictive or less restrictive?

Answer 9: The Agency has reviewed the Sandia Report, and has noted the potential updates. However, at this time, the Agency has not made decisions on its direction for these issues, and as a result, will determine the need for further technical analysis after comments are received from the public comment period for the ANPR.

Question 10 (addressed to Brian Littleton, EPA): The US EPA was relatively silent following the identification of tritium release at Braidwood. Please explain the role of the federal EPA for these types of releases from nuclear power site.

Answer 10: The EPA enforcement actions for ground water contamination are led by the particular region the contamination is suspected. In the event additional support is needed beyond the regions capability, HQ staff and resources can be committed. Decisions on enforcement are made dependent upon the level of contamination made available to the public. The State also plays an important role in determining the appropriate level of response. In the event the contamination has exceeded a regulatory limit, then the Region will make decisions on the best course of action for proceeding. In the case of the Braidwood contamination, data indicated that contamination levels in private wells did not exceed the drinking water MCLs.

Question 11 (addressed to Brian Littleton, EPA): What are main considerations on spent fuel storage radiation protection?

Answer 11: The Agency wants to clarify that all spent fuel storage operations are covered by an environmental protection standard, and clarifying which radiation protection standards apply to each situation identified. It is also important to understand the implications of extended storage to assess the adequacy of the current standards.

Question 12 (addressed to Brian Littleton, EPA): How does the linear non- threshold model- which is arguably not best estimate- comport with EPA's risk- based decision making framework- which is based on a best- estimate understanding of epidemiology?

Answer 12: EPA's radionuclide risk estimates are consistent with the LNT approach. EPA's risk-based framework seeks to limit risks from man-made radiation to a risk number (generally 10-4 to 10-6). The LNT model assumes that for every dose from man-made radiation, there is a proportional risk in developing cancer.

Question 13 (addressed to Brian Littleton, EPA): Coordination between EPA and NRC to ensure alignment of regulations is critical, but specifically not discussion. What level of coordination exists, and why not more? Comment: one regulator/ one responsible enforcement agency.

Answer 13: Same as question 4. The EPA and the NRC have held more frequent meetings to keep each agency appraised of potential changes as we move forward with revisions to 40 CFR 190, 40 CFR 192, and potential changes to the NRC's radiation protection regulations. Although different statutory drivers may necessitate minor differences, the hope is to minimize these differences, to the degree possible.

***TH34** – Regulatory Actions Regarding Containment Venting and Filtration*

Session Chair: William Ruland, NRR

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

***TH35** - Software-Based Tools for the Development of Digital Safety Systems*

Session Chairs: Steven Arndt, NRR, Daniel Santos, NRO

Session Coordinator: Stephen Wyman, NRR
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The questions below were not answered during the above session:

Question 1: Is 95% / 95% an adequate concept in the IVV of software Tools? Is it possible through the analysis techniques described to achieve a 95/95 state of confidence / likelihood of success in assuring no fatal flaws?

Answer 1: Statistical methods for confidence and likelihood are normally applied to safety systems whose accuracy and reliability can be easily quantified. For computer based systems, this has been difficult to achieve and therefore these statistical methods have not been applied. The methods discussed for evaluating software tools are intended to provide a level of assurance that the tools will not introduce errors, into the safety related code being developed, that cannot be detected by verification and validation activities.

Question 2: Richards point is that "tools can introduce failures" while Connie's point is "without tools you won't find the failures". So, why is the NRC not insisting on the use of tools?

Answer 2: While the NRC encourages the use of software tools when they provide benefits to the software development process, it is not the NRC's role to endorse or dictate the use of any particular method of software development. Vendors should choose tools based on the benefits that they provide to the software life cycle process being used. They should also develop processes that ensure that all tools are used correctly and within their design limitations.

Question 3: "Safety Related" has been mentioned frequently. How are you determining the safety significance of the software or parts of the software?

Answer 3: The safety significance of software is determined by what the software is being relied upon to accomplish. In general safety related software is designed to perform safety functions such as; initiate safety injection or trip the reactor. Therefore, this type of software should be classified as having high safety significance. Software tools, on the other hand do not directly perform any safety related task so they are often classified as having low safety significance. However, we also consider the roles that software tools have in the development process. Depending on how a tool is used and what it is being used for, there may be a significant reliance on the correct performance of a tool to produce reliable safety significant software. In these cases it would be appropriate for the tool to be rated at a higher level of safety significance.

Question 4: NUREG/CR-6303 was mentioned as guidance on diversity. What do you think of the new document NUREG/CR 7007 that tries to answer how much diversity is enough?

Answer 4: NUREG/CR-7007 provides a means of objectively analyzing and quantifying the level of diversity that exists between systems. Defining the level of diversity still does not answer the question of; is the level sufficient, which remains a judgment call on the part of the reviewer. NUREG/CR-7007 is a good tool that provides useful information to help us to make more informed decisions on adequate diversity.

Question 5: When would a licensee be required to submit software tools documentation to the NRC for approval?

Answer 5: The Standard Review Plan contains guidance for the staff on how to review and evaluate software tools in conjunction with a safety evaluation. If the information needed to complete this assessment is not included in a license amendment, then the staff will likely ask for the information in order to provide a basis for the safety conclusions. Refer to Standard Review Plan, Chapter 7.1-D and Regulatory Guide 1.168 for more information on this topic.

Question 6: Why does the NRC not just endorse the software tools guidance from IEC 60880?

Answer 6: Though there is a significant amount of overlap between IEC standards, and the IEEE standards that the NRC staff endorses, one can always come up with a way to meet one without complying with the other. The criteria of either standard are not encompassed by the other. We are working with the international community and with IEEE to close this gap.

Question 7: Wouldn't a software tool that has been previously evaluated to be in compliance with IEC 60880 also be in compliance with IEEE 7-4.3.2 and thus be acceptable to the NRC?

Answer 7: Not necessarily. There are criteria within IEEE 7-4.3.2 for which the IEC standard has no equivalent criteria. It is possible for a tool to be compliant with both though, but you would have to perform two separate evaluations to confirm this.

Question 8: Can you give an example of a method that can be used to verify that a software tool is performing correctly?

Answer 8: Yes. Using the spreadsheet scripting example from my (Rich Stattel's) presentation, you could test the script before and after each test by loading a tool test data set and confirming that the script produces the correct and consistent output.

Question 9: What does Westinghouse do to ensure that the tools delivered to the licensee are being controlled and used correctly at the plant?

Answer 9: The licensee is responsible for having their own configuration management plans and procedures for software resident at the NPP, including tools.

Question 10: What kinds of controls or qualifications would you expect to see for test support tools that are highly configurable such as Labview?

Answer 10: The answer to this depends on what the tool is being used for and how it is being used. A tool analysis should be performed to identify each of these parameters and the results of this analysis should include direction on how to control the software configuration and usage.

Question 11: Do you have the explanation or reasons for what failed on the tragic France Airbus plane that fell into the ocean and killed all aboard last summer?

Answer 11: Air France Flight 447 crashed on June 1, 2009. The official report is still pending. Unofficially, it has been reported, based on black box data recovered in May, 2011, that inconsistent air speed sensor readings and pilot error caused an aerodynamic stall.

Question 12: In actual NPP construction process, the instruments and components related to the safety I&C system are selected or finalized at the later stage of construction, which affects the safety software development and verification. This kind of software change is time consuming and usually has a big impact on the whole project schedule. Is there any smarter way to handle this situation which is acceptable both regulator and software developer, i.e., utility?

Answer 12: Early on in the design process, Westinghouse defines the design requirements for the sensors and actuators. These requirements are passed down to vendors as purchasing requirements. This allows the I&C design to be finalized before construction.

Question 13: Is the I/O, General Purpose software considered safety related per NRC safety related requirements?

Answer 13: The General Purpose software category in the Common Q Software Program Manual is not considered safety related. I assume the questioner is asking if the I/O Simulator software is considered safety related. The answer is no.

Question 14: With Westinghouse's categorization of its validation software tool as "General Purpose" software (versus the more conservative and rigorous Westinghouse categories of "Important to Safety" and "Protection"), does it satisfy Method 1 or Method 2 in 7-4.3.2 as discussed in the slide by Rich Stattel?

Answer 14: The current regulatory guidance is Regulatory Guide 1.152, Revision 3 which endorses IEEE 7-4.3.2-2003. IEEE 7-.4.3.2-2003 states the following in regards to software tools:

One or both of the following methods shall be used to confirm the software tools are suitable for use:

a) *A test tool validation program shall be developed to provide confidence that the necessary features of the software tool function as required.*

b) *The software tool shall be used in a manner such that defects not detected by the software tool will be detected by V &V activities.*

The I/O Simulator software was subjected to a validation program in accordance with method a). In addition, the I/O Simulator was developed using a high quality process requiring a system requirements specification and software design specifications (i.e. the design process called out for General Purpose software in the Common Q Software Program Manual).

TH36 *- The Safety/Security Interface at Nuclear Power Plants—an International Interest*

Session Chair: Marc Dapas, NSIR

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The questions below were not answered during the above session:

Question 1: Does the NRC see a future rule for security plans similar to 10 CFR 50.54(q) change for emergency plans to prevent a reduction in effectiveness?

Answer 1: The NRC does not see the need for such a rulemaking. Under 10 CFR 50.54(p)(2) a licensee may make changes to its security plans, without prior Commission approval, if the licensee determines that such changes do not decrease the safeguards effectiveness of the plan. The licensee is required to submit a report of such changes within two months of the change. Additionally, the licensee must maintain records of such changes for a period of three years after the date of the change. The NRC may elect to inspect those records.

Question 2: Would it be correct to include safeguards into the synergy of “3-S concept,” safety, security, and safeguards? If so, how are these interfaces accounted for in the IAEA guidance?

Answer 2 (response from IAEA): A synergetic area where nuclear security and safeguards support each other is the system for nuclear material accountancy and control (NMAC) at the facility level. NMAC at the facility level as a security component, coordinated with the physical protection, is designed to maintain an inventory of all nuclear material, including information related to its quantity, type, location, use, movement, and transformation of all nuclear material; and to register an alarm if an anomaly is detected. In addition, safeguards agreements and protocols, concluded between the IAEA and States, provide, inter alia, through state systems of accounting for control of nuclear material, central contributions to preventing illicit trafficking, and to deferring and detecting the diversion of nuclear material. In 2012, the Agency started a project on development of an action plan and technical guidance for synergy between safety and security. It is also clear that technical guidance on 3-S interface would be developed by IAEA in the future.

Question 3: How safety-security interface issues are addressed by the NRC for fuel cycle facilities as opposed to power reactors? In particular, how are theft concerns addressed for CAT I facilities?

Answer 3: NRC verifies compliance with 10 CFR 73.58, safety-security interface requirements through its baseline inspection program for operating reactors. The CAT I facilities inspection program is similar to that of the operating reactors program.

Question 4: The NRC has been a big proponent of safety-security as evident in 10 CFR 73.58. Are other U.S. agencies and the international community adopting these concepts?

Answer 4: The NRC routinely cooperates with the Department of Energy and IAEA in the area of safety and security to ensure proper use and protection of nuclear materials. NRC rules and regulations, including safety-security interfaces, are communicated with these agencies to clarify roles and responsibilities. As such, the safety and security interface concept and the need to consider this important area is being considered in the development of IAEA Nuclear Security Series documents and guidance. The NRC also engages in a number of international training courses in physical security protection to make awareness of the importance of safety security interface and its consideration in the regulatory infrastructure development.

Question 5: Developing security guides and training in the form of open-public documents and promoting training courses on those subjects have a potential to give access to terrorists or malevolent people with sensitive information that may help them facilitate malevolent actions. How do you intend to conciliate such actions with real security confidentially? At least a strict control of either the content of guides/training, or of the people who have access, or can attend courses have to be put in place. (Panel)

Answer 5: NRC develops and conducts physical protection training courses to support international regulatory counterparts to communicate methodologies and processes in physical security protection. These methodologies and processes are best practices and guidelines. Detailed implementations of these processes and guidelines are controlled security-related information and are not made publicly available.

Answer 5 (response from IAEA): To avoid any misunderstanding, we should define that part of nuclear security information which should be treated as sensitive or confidential. It is, for example design features and technology of construction of an improvised nuclear explosive device of an RDD (“dirty bomb”). Another examples are exact content of security plan of a nuclear facility, number of facility guards and their arms, design based threat details or vulnerabilities of a concrete security system. This information is the property of States and their nuclear operators and regulators, not the IAEA. As for the IAEA nuclear security publications and training materials, we provide recommendations and guidance as security measures, which are applicable to different nuclear and other radioactive materials and the associated facilities and activities. The IAEA guidance documents present best practices that should be or could be adopted by Member States in the application of the nuclear security objectives and principles. People, attending IAEA human resource development activities or contributing to the establishment of IAEA guidance are nominated by States.

Question 6: Is there an integrated methodology for simultaneously assessing safety and security, both challenges and outcomes? (Panel)

Answer 6: Safety-security interface is part of security physical protection baseline inspection activities. The safety-security interface inspection is conducted every 2-year cycle. Inspection activities focus on licensee’s workflow and review programs and communications across organizations on those configuration changes that could simultaneously affect both safety and security objectives and effectiveness.

Answer 6 (response from IAEA): Integrated methodology of simultaneous assessment of nuclear safety and security does not exist yet. The challenge is use of different assessment tools: design based accident for safety (e.g. range of conditions and events exceeding authorized limits of a safety system) and design based threat for security (e.g. attributes and characteristics of adversaries with malicious intention). We consider, inter alia, opportunities to integrate, where justified, both methodologies for safety and security assessment in relation to NPP to coordinate evaluation of consequences of a safety accident with consequences of a radiological sabotage action.

Question 7: How do you minimize the insider threat, when integrating operations into security issues?

Answer 7: 10 CFR Part 73 requires that licensees have a physical protection plan and system to protect against acts of radiological sabotage and to prevent the theft or diversion of special nuclear material. The requirements are intended ensure that individuals are reliable and trustworthy. In particular, 10 CFR 73.56 and 73.57 require that individuals granted unescorted access must meet the personnel access authorization requirements and satisfy the criminal history records checks to ensure that they do not constitute an unreasonable risk to public health and safety, or the common defense and security, including the potential to commit radiological sabotage.

Answer 7 (response from IAEA): Operations with sensitive nuclear material in inner areas of a nuclear facility and in vital areas of nuclear installations of high radiological consequences (NPP) should be arranged by application of strong security measures like minimization of access point to these areas, provision of access to minimum necessary personnel and keeping records of all of them, "two-person rule," detection and delay of unauthorized actions within the areas.

Question 8: Mr. Brunt's (UK) presentation included a suggestion that safety and security interface can extend beyond a plant's operational life, all the way to decommissioning. Do NRC regulations and IAEA's guides include consideration of plant decommissioning? (Panel)

Answer 8: 10 CFR 73.58, safety and security interface is applicable to operating reactor licensees pursuant to 10 CFR 50.22 when fuels are onsite.

Answer 8 (response from IAEA): IAEA nuclear security recommendations and guidance is applicable to all stages of a nuclear facility construction, commissioning, operation and decommissioning. For example, risk of unauthorized removal of nuclear and other radioactive material of spent fuel pool exist during the decommissioning process of a nuclear facility as long as the material is not removed for disposal from the facility site.

Question 9: The cyber security rule (Part 73.54) requires significant safety-security interface. How is the control of critical digital assets being addressed with regard to safety and security?

Answer 9: To ensure that the cyber security program does not adversely impact the proper functioning of critical digital assets (CDAs), Regulatory Guide 5.71 "Cyber Security Program for Nuclear Facilities," states that security controls should only be applied to CDAs if the implementation of the security controls does not adversely impact a CDA's functions or performance. Therefore, if implementation of a security control on a safety system is determined to have an adverse effect on the safety system's performance or functions, the security control should not be applied. Instead, an alternative security control that does not impact a safety system's function or performance should be applied to protect that safety system from cyber attack, up to and including the DBT.

Question 10: What qualifications and training are required for the newly assigned Security Work Control and Projects Analyst position? (Exelon)

Answer 10: The position description for this position in Exelon relates that the person applying for this position will be qualified Armed Security Officers with at least 5 years experience. The preferred candidate would have Work Management experience and have advanced to the "Security Crew Lead". This denotes detailed knowledge of the Site Security Plan, Target Set Equipment and proven ability to interact well with other department personnel. All essential to the successful execution of duties associated with this position.