SESSION QUESTIONS AND ANSWERS

Below are questions and answers which were not addressed during the technical session portion of the conference.

William D. Magwood, NRC Commissioner, "Regulating a Renaissance: The NRC's Role in Globalized, Environmentally Conscious, Security-Focused, and Economically Uncertain Century"

Question 1: Many attempts were made to communicate the EPA drinking water standard and basis at Braidwood. I find it fascinating that you sensed that the members of the local public did not understand this.

Answer 1: I was surprised as well. It is possible that the fact that the staff's many previous efforts to address the EPA drinking water standard to the Braidwood community appear to have been not entirely successful points to the need to take a close look at how we as an agency undertake efforts to communicate scientific and technical information to the public. I encourage the staff to begin a dialogue, both internally and with outside stakeholders, regarding how we can better communicate complex ideas so that we can ensure that, going forward, interested members of the public have full, accurate, and credible information and background on the issues.

Question 2: Your comment about "it takes as long as it takes" is quite disturbing, relative to predictable and reliable regulation. Is there not some responsibility on the part of NRC to move issues to closure on a reasonable schedule, rather than endless input and debate?

Answer 2: Absolutely, and I believe we, in large respect, strike an appropriate balance. But we must always recognize that NRC's first mandate is protection of public health and safety. Therefore, there may be situations where the complexity of a technical issue requires the NRC to deviate from its established schedule in order to ensure that its safety review is complete and thorough and will ensure the adequate protection of public health and safety. Although the NRC will always strive to meet its established schedules and milestones, the NRC must never elevate schedule over safety.

Dr. George Apostolakis, NRC Commissioner, "The First Year"

Question 1: Will NRC independently investigate allegations and reports of harassment and wrongdoing at NRC licensee Oregon State University Nuclear Engineering Department?

Answer 2: Members of the public or people working in NRC-regulated activities may report safety concerns directly to the NRC by discussing the issues with an NRC staff member, calling the NRC's Safety Hotline 800-695-7403, or writing a letter to the NRC. A brief summary of the allegation program can be found in our brochure on Reporting Safety Concerns to the NRC (NUREG/BR-0240).

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Session Day and Time: Tuesday, 1:30 pm–3:00 pm

Session Number and Title: T2 Current Activities and Future Plans for Small- and Medium-Sized Reactors

Session Chair: William Reckley, Branch Chief, Division of Advanced Reactor Program, NRC/NRO

Session Coordinator: Wesley Held, NRC/NRO, tel: (301) 415-1583, e-mail: Wesley.Held@nrc.gov

Question 1: On legal changes. Have you identified which changes will require Congressional action and which will only need NRC action?

Answer 1: Discussions between stakeholders, DOE and the NRC have not identified issues that would require legislative changes. Evaluations continue on matters such as liability insurance (Price Anderson Act) but it has not been concluded that legislative actions are needed.

Question 2: What are the waste heat removal methods?

- open cycle once through water or
- closed cycle

Answer 2: The waste heat removal methods will be determined based on various design and site factors. A possible advantage for smaller units is that air cooled condensers are a possibility and this could address some concerns regarding water usage.

Question 3: What are the water usage strategies for makeup?

Answer 3: The waste heat removal methods will be determined based on various design and site factors. A possible advantage for smaller units is that air cooled condensers are a possibility and this could address some concerns regarding water usage.

Question 4: What industry activity would best complement IAEA work on SMRs?

Answer 4: Participation in IAEA discussions, working groups and other activities would probably be the most useful step that the US industry could do. We are developing various standards, technical papers, and studies regarding the use of nuclear energy in various global markets and so input from various plant designers is very useful.

Question 5: Please say more on MDEP "umbrella" initiative.

Answer 5: The MDEP programme incorporates a broad range of activities including:

- Enhanced multilateral co-operation within existing regulatory frameworks.
- Multinational convergence of codes, standards and safety goals.
- Implementation of MDEP products to facilitate licensing of new reactors, including those being developed by the Generation IV International Forum.

A key concept throughout the work of MDEP is that national regulators retain sovereign authority for all licensing and regulatory decisions. For more information, visit http://www.oecd-nea.org/mdep/

Question 6: For transportable nuclear power plants on barges, what would happen in case of severe storms? Would the barge leave the dock and move to open water or would the plan be to ride out the storm in a well-protected harbor? What implications does this have for siting or operations?

Answer 6: Most transportable designs are intended to be moved to a location and then generally become a "stationary" power source for a period of time. This means that the plant design, site and interface features such as docks need to meet established criteria regarding external hazards.

Question 7: How will costs for security, EP, and other fixed programs affect SMR power delivery costs?

Answer 7: The goal of SMR designers is to as much as practical address security and other requirements that add to operational costs in the design of the facilities. While not eliminating these costs, it is hoped that the operational costs might be reduced by taking such matters into consideration during the design.

Question 8: Has the President or the Secretary recognized officially that nuclear power is to be considered equally with renewable energy sources (wind, solar, etc.) when providing federal funding?

Answer 8: Both the President and Secretary of Energy Chu have stated that nuclear energy should be part of the country's energy mix going forward and will play an important role in addressing concerns about greenhouse gas emissions.

Question 9: Since when does "eliminating local hiring unions" equate to "high quality jobs"?

Answer 9: While the DOE does not make decisions regarding the labor arrangements for commercial nuclear plants, we have performed assessments of the existing infrastructure for related construction projects and found shortages in many areas of the country. A finding was that engineering companies should consider negotiating a national labor agreement with major labor unions to provide flexibility in staffing nuclear construction projects (e.g., allowing union members from different areas to work at any nuclear plant construction site). This step would help ensure the needed construction workers will be available.

Question 10: Your "Path Forward" included "supporting development of new standards" for SMRs. Are you referring to voluntary consensus standards? What does "support" mean? Do you have funds to help expedite standards development?

Answer 10: The DOE is providing financial support and working with the National Institute of Standards and Technology (NIST) as well as with standards setting organizations (e.g., American National Standards Institute) regarding the updating of nuclear related industry codes and standards.

Question 11: If the 1st SMRs are DOE- or DOD-owned, how will NRC regulation be incorporated so to assure commercialization?

Answer 11: The current planning does not call for DOE or DOD to own or operate the lead SMR plants. Instead, the DOE could cost share some of the development and licensing and DOE/DOD could be a customer for the electricity generated. This logic was discussed in the "push/pull" slide in my presentation.

Question 12: When you talked about interagency cooperation you did not mention EPA, which would clearly have a very large role to the extent SMRs start becoming a popular alternative. What are the pros & cons environmentally speaking, specifically fuel?

Answer 12: The DOE and NRC will be coordinating activities with other federal agencies including the Department of Homeland Security, the Federal Emergency Management Agency, Environmental Protection Agency and others. The environmental reviews will depend on the technologies being discussed but possible differences include the possibilities of reduced water use, the replacement of older fossil fueled plants, and a smaller overall facility footprint.

Question 13: Have you looked at fueling the SVBR with thorium? What would be the issues with thorium?

Answer 13: Fueling the SVBR with thorium is under consideration on conceptual level as well as the rather effective approach especially for such countries like India.

Question 14: Has the NRC thought about any changes to emergency planning or are you waiting for industry input/ideas first?

Answer 14: The topic of emergency planning for SMRs has been discussed at several public meetings between the staff and stakeholders, including a meeting on a white paper submitted by the next generation nuclear plant program. The staff is preparing a Commission paper and expects to further engage stakeholders regarding possible approaches.

Question 15: SMRs are being discussed as replacements for coal plants. Some of these are located near population centers. What are the NRC's plans for reviewing requirements for EPZs?

Answer 15: The topic of emergency planning for SMRs has been discussed at several public meetings between the staff and stakeholders, including a meeting on a white paper submitted by the next generation nuclear plant program. The staff is preparing a Commission paper and expects to further engage stakeholders regarding possible approaches.

Question 16: How is the NRC anticipating the safety program/design of small reactors in comparison with traditional large reactors? Will it have similar requirements, for example, the definition of safe shutdown earthquake as far as the probability of exceedance?

Answer 16: In general, the same requirements will apply to the SMR designs as has been used for the recent large reactor designs. The NRC staff and plant designers are currently assessing regulations, guidance documents, and other material to determine applicability to the specific SMR designs. The results of these assessments might be that a requirement or guidance document is applicable as is, not applicable, applicable but requires some revision, or that a new requirement/guidance document needs to be developed.

Question 17: Has the NRC evaluated the use of fossil fuel capital equipment manufacturers to launch an initiative or partnership to produce both clean coal and SMR components?

Answer 17: It is the responsibility of the reactor vendors, with possible support from DOE or other agencies, to develop the plans and infrastructure for the fabrication and construction of

SMRs. It has been mentioned that US domestic manufacturing capabilities may be better suited to SMRs than has been the case for large reactors. The NRC expects to evaluate reactors and other systems, structures and components; component vendors; and other contributors to reactor equipment using its existing oversight and inspection programs.

Question 18: What is the relationship between the efforts of NEI & ANS in the area of SMRs?

Answer 18: The American Nuclear Society Special Committee on Generic Licensing Issues for SMRs is preparing a series of white papers. Some of the issues being addressed by the ANS are also the subject of NEI position papers. The NEI efforts are informed by the ANS work, the next generation nuclear plant (NGNP) activities, and our own evaluations. In some cases, the NEI papers are built extensively on these earlier efforts.

Question 19: Is there any work on international deployment issues envisaged by the NEI TF or WG? What along this line would be beneficial?

Answer 19: NEIs efforts are presently focused on domestic deployment of SMRs. Therefore we are focused on how SMRs fit into the existing NRC regulatory framework. However, we will certainly monitor international activities for lessons.

Question 20: Should Price-Anderson apply to SMRs? What about private insurance?

Answer 20: NEI currently has a working group evaluating liability insurance issues, including the provisions defined by the Price Anderson Act. A position paper on this topic is expected to be submitted to the NRC in the near future.

Question 21: Should the automatic proximity standing for those within 50 mile radius be eliminated for SMRs? It is already not applicable for materials licensees.

Answer 21: The NEI working group is not currently evaluating the issue of NRC hearings and related matters such as the standard 50-mile criterion used standing determinations.

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Session Day and Time: Tuesday, 1:30 pm–3:00 pm

Session Number and Title: T3 Digital Instrumentation and Control Research: Strengthening Regulatory Decisions

Session Chair: Russell Sydnor, Branch Chief, Division of Engineering, NRC/RES

Session Coordinator: Milton Concepcion, NRC/RES, tel: (301) 251-7457, e-mail: Milton.Concepcion@nrc.gov

Question 1: What has been done to quantify digital I&C systems failures (software and hardware)?

Answer 1: The NRC Division of Risk Analysis (DRA) in the Office of Research (RES), working with Brookhaven National Labs, is investigating methods for quantification of digital I&C system failures. RES/DRA is planning an update report to ACRS on this work in June 2011. Obtaining data that supports system failure quantification remains a challenge. The Office of Research intends to obtain updates from EPRI under a MOU on EPRI research in digital system failure analysis.

Question 2: Is it reasonable to use dynamic PRA on software but not hardware? In other words, is it a must that they have to be done together?

Answer 2: There is no consensus agreement on the treatment of digital system software and hardware in PRA. The NRC has investigated several potential methods for digital system PRA, including dynamic methods, through research activities (NUREG/CR's - 6942, - 6985, - 6962, - 6997). ACRS review of the research results from a study of traditional PRA methods for use in a digital system PRA questioned that the study did not include software as well as hardware. Determining the software failure modes is a very challenging concept that in fact may not be meaningfully achievable. There may be alternative methods that could be used for estimating digital system reliability for use in a PRA and understanding the uncertainty of such a digital system PRA.

Question 3: Could each member expert panel raise their hand if they believe that digital control systems are <u>more</u> reliable in nuclear non-safety applications?

Answer 3: The NRC is not aware of any study comparing the reliability of digital versus analog control systems in nuclear non-safety applications.

Question 4: After hearing the "don't have yet" list from Dr. Wassyng, how can the NRC conclude "adequate assurance of safety" in the NRC reviews?

Answer 4: Assurance of safety is the responsibility of the licensee. NRC licensing reviews depend upon the integrity if the applicant's claims, the evidence available for review, and judgment. As the panelists, including Dr. Wassyng, stated in the RIC DIC session, sysem complexity is the most dominant factor affecting assurance. While simpler systems of the past were verifiable with higher confidence, as complexity increases, uncertainties increase and confidence drops. Dr. Wassyng identified the gaps to be filled, in relation to the complexity seen nowadays.

Question 5: As we separate safety from control, do we have a way to separate software failures from hardware failures?

Answer 5: Interpreting the use of the word, "separate" to "prevent adverse effect of" one on the other, the answer is "yes." To prevent adverse effect of one component on another, the system should have appropriate architectural constraints and provisions, proper selection of hardware components, and thorough verification and validation. For example, although a hardware component, subjected to wear and tear, is likely to fail eventually, only those components should be selected, for which the fault modes are well known, well understood, degradation is graceful (rather than sudden), and faults are detectable independently early enough. Then, independent monitors should be applied to detect the fault before system failure, and actions should be provided to bring the system to a safe state. Thus, the architecture should prevent the propagation of the effect of a fault in one hardware component to the rest of the system. In the case of software, the architecture should prevent its corruption during operation and prevent the propagation of the effect of a fault in one component to the rest of the system. Thorough verification of each software component should assure that it is free of latent defects. Simplicity is a key property of the system to achieve these properties.

Question 6: The presenter had stated that enhanced reliability does not mean increased plant safety. Previously the NRC has accepted that increasing reliability does enhance safety by reducing transients caused by spurious or failed signals. Has the NRC changed its viewpoint?

Answer 6: The staff has not changed its viewpoint on this matter. However, during the presentation, the staff has emphasized that adding functions to support a slight increase in reliability may introduce additional failure modes that could adversely impact the safety function. Therefore, the designer should balance between functions that support an increase in reliability with the effects that these functions can adversely impact safety functions.

Question 7: Can the presenter provide 1) an example of a case where data sharing among redundant divisions are found to be acceptable and 2) a case where data sharing among redundant divisions were found to be unacceptable, and why?

Answer 7: An example to support interdivisional communication is the use of interdivisional data sharing to support multi-divisional display and control of certain plant variables and equipment if the human factors analysis demonstrates a significant benefit to the operator for such functions. Otherwise, if the analysis shows that such functions do not provide benefit to the operator, then data sharing among redundant divisions may not be acceptable.

Question 8: NRC's approach to simplicity negates all the beneficial safety enhancements offered by digital I&C systems. Why does the NRC not value the benefits from signal validation and increased reliability, and improved HMI display capability?

Answer 8: The NRC staff did not advocate that digital I&C systems should not be implemented for nuclear power plants. However, the staff emphasized the importance of ensuring that the implementation does not adversely impact the safety function. This can be achieved by ensuring that fundamentals principles of independence and redundancy are maintained by whatever design approach is taken.

Question 9: Does a design that satisfies the single failure criterion through means other than redundancy and independence satisfy the intent of GDC 21?

Answer 9: GDC 21 requires that redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the safety function and (2) removal from service of any component or channel does not result loss of required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. Therefore, the requirement of GDC 21 is to have single failure protection and the capability for test and maintenance without loss of minimum redundancy. GDC 21 implies that redundancy and independence are the preferred means to address the two criteria above, but other means to address the criteria may be acceptable. Note that 10 CFR 50.55a(h) does require independence through Clause 5.6 of IEEE Std. 603-1991.

Question 10: Dr. Wassyng stated that V&V etc. must be performed during the development process to be effective. For the NRC review process to be effective, shouldn't the NRC review process also follow that maxim? That is, should the NRC review be done during the development rather than after the system has been developed?

Answer 10: The NRC staff reviews the applicant's design and the development process as part of the review of a digital safety system to verify whether the design meets regulatory requirements and that the process is of sufficient high quality to produce systems and software suitable for use in safety-related applications in nuclear power plants. Further information can be found in Interim Staff Guidance 6, "Licensing Process."

Question 11: Regarding forensic tools, are you suggesting that regulators purchase the tools, train on them, and use them when reviewing licensing?

Answer 11: Forensic tools were discussed during the presentation as a way to facilitate the verification and certification of software. The presentation also mentioned that additional work needs to take place to develop safe and effective forensic tools. The NRC is conducting research to explore alternative assurance methods that may be considered for future use by the staff during the review of digital safety systems.

Question 12: How do you account for software modules and operating systems not developed in-house?

Answer 12: Regulatory requirements and guidance are applicable to digital safety systems, which includes software and hardware products developed by third parties. Licensees are responsible for the design, procurement, and quality assurance of safety systems, whether they are developed in-house or by external vendors.

Question 13: Could analog systems be more reliable than digital systems due to their relative simplicity? If so, why not use analog for key safety systems and use digital for non-safety and monitoring of safety systems?

Answer 13: NRC regulations and guidance on safety system do not specify the use of analog or digital systems in a plant application. It is the licensee option to select the safety system design and the staff reviews the regulatory adequacy of the proposed design.

Question 14: What is different about how to apply failure modes and effects analysis tools when evaluating new digital functions?

Answer 14: FMEA plays a role in the design and development of safety systems, including digital. Recent research has determined that the use of FMEA for software assurance is of limited value; however, practitioners have found that software FMEA can be useful as part of the software hazards analysis in the development process.

Question 15: Recognizing how digital components fail is addressed in most applications. However, equally important is how digital components respond upon repowering after loss of power. How is this being addressed in regulatory space?

Answer 15: Digital safety systems are required to perform their safety function when subjected to conditions, external or internal, that have significant potential for defeating the safety function. Additional guidance are found in RG 1.152, which endorses IEEE 7-4.3.2, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations."

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Session Day and Time: Tuesday, 1:30 pm–3:00 pm

Session Number and Title: T4 NRC Enforcement Policy—Where We Have Been and Where We Are Going

Session Chair: Roy Zimmerman, Office Director, NRC/OE

Session Coordinator: Kerstun Day, NRC/OE, tel: (301) 415-1252, e-mail: Kerstun.Day@nrc.gov

Question 1: "Last fall, the NRC issued a white finding to the Robinson licensee for the most miniscule problem with an emergency diesel generator. Why was this white finding issued for a problem that would not have even produced a green finding anywhere else?"

Answer 1: The question apparently refers to a White Finding issued December 7, 2010. Specifically, the NRC concluded that the failure to correct an emergency diesel generator (EDG) output breaker failure in October 2008 constituted a performance deficiency which resulted in the inoperability of the EDG in April 2009. The NRC determined that the first breaker failure, including the specific nature of the symptoms observed by maintenance personnel, provided a reasonable opportunity to identify and pursue correction of the breaker failure mode and therefore prevent the inoperability of the EDG the second time. Following the two occasions when the breaker failed to close in October 2008, the licensee did not take corrective actions commensurate with the safety significance of the equipment or the narrow nature of the symptoms, as evidenced by the return of the breaker to service without performing thorough troubleshooting and without attempting to determine the exact cause of its failure. The NRC concluded that had licensee personnel dedicated the same level of effort and attention to the first breaker failure in accordance with the plant's corrective action program, as they did for the second failure approximately six months later, the inoperability of the EDG for over 3 weeks could have been prevented. The basis for the statement "issuing a White finding for an issue that would not have even produced a green finding anywhere else" is unclear. The Reactor Oversight Process (ROP) uses a significance determination process (SDP) which considers the plant specific aspects when determining the risk significance of each issue. When processed through the SDP, the facts of the Robinson EDG issue, combined with the plant specific information identified that the issue was of low to moderate safety significance and was appropriately designated as a White finding.

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Session Day and Time: Tuesday, 1:30 pm–3:00 pm

Session Number and Title: T5 Radiation Protection

Session Chair: Stephanie Bush-Goddard, Branch Chief, Division of Systems Analysis, NRC/RES

Session Coordinator: Vered Anzenberg, NRC/RES, tel: (301) 251-7546, email: Vered.Anzenberg@nrc.gov

Question 1: How do challenges to the linear no threshold (LNT) model affect or not affect work at each agency?

Answer 1: The NRC staff is carefully monitoring the ongoing research on the effects of low doses of ionizing radiation. This includes interactions with the Department of Energy Low Dose Program, and work done domestically and internationally by organizations such as the United

Nations Scientific Committee on the Effects of Atomic Radiation, the International Commission on Radiological Protection (ICRP), the National Council on Radiation Protection and Measurements (NCRP), and the National Academies of Sciences. The current level of scientific understanding does not provide specific information on the dose to effect relationship in the low dose area. In the absence of peer reviewed information that would be generally applicable to the protection of humans, the Nuclear Regulatory Commission continues to use a linear no threshold model for regulation, as recommended by the ICRP and NCRP.

Question 2: What new or proposed radiation guidelines could affect nuclear power plant decommissioning procedures and/or costs?

Answer 2: The Commission has recently completed work on regulatory changes to decommissioning planning, which should be published in the Federal Register in the near future. As the staff continues its interactions with stakeholders on the possible benefits and impacts of increasing alignment of NRC's regulatory framework with international recommendations, it will continue to consider if changes may be appropriate. At this time, stakeholders have not indicated a need for, or scientific information supporting a change to the radiological criteria for decommissioning.

Question 3: Given the draft policy statement which indicates how strong the security for these sources, why is the NRC proposing so many new requirements in part 37?

Answer 3: The Draft Policy Statement addressed the issue of security considerations for cesium-137 sources in broad terms. The specific provisions of the proposed 10 CFR 37 regulation are outside the scope of the Draft Policy Statement. Questions about the proposed regulation should be addressed to the contact person for the proposed regulation, Ms. Merri Horn, telephone: 301 414-8126, e-mail: merri.horn@nrc.gov.

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Session Day and Time: Tuesday, 1:30 pm–3:00 pm

Session Number and Title: T6 Regulatory Impacts of International Operating Experience

Session Chair: Michael Cullingford, Special Assistant for Technical Policy and International Liaison, NRC/NRR

Session Coordinator: Michael Cullingford, NRC/NRR, tel: (301) 415-1276, e-mail: Michael.Cullingford@nrc.gov

Question 1: What is NRC policy regarding the use of INES Scale? What is the yearly average of events classified above Level 0?

Answer 1: The NRC reviews every Event Notification submitted in accordance with 10 CFR 50.72 to determine the appropriate INES rating. Events that are determined to have an INES rating of level 2 or higher are submitted to the IAEA NEWS website. Typically, out of about 400 reactor events received each calendar year, between five and ten are rated level 1, and the rest are either level 0 or do not fall within the scope of the scale. The last level 2 event prior to the HB Robinson event was in 2005.

Question 2: Re: HB Robinson Event: What changes, if any, are expected to the ROP Program and event screening for non-safety related systems, like those that impacted the HB Robinson Event, with the goal of preventing future events?

Answer 2: Although the fault initiated in a non-safety related cable, the primary complications experienced at HB Robinson arose from process and procedural deficiencies. No major changes to the ROP are seen as being necessary following this event. Within the scope of the ROP, an Operating Experience Smart Sample providing focused inspection guidance addressing licensee simulator training and an Information Notice informing industry of one particular issue which significantly complicated plant and operator response to the initial fault, were issued in the immediate aftermath of the event. Following more in-depth review by the Augmented Inspection Team, identified performance deficiencies were addressed and will result in increased inspections at the site. A follow-on generic communication is expected which will provide further notification to industry of the sequence of events, their consequences, and lessons-learned.

Question 3: Does the NRC have plans to internationally promote interactions and exchange among "Clearing Houses" such as the European Cleaning House or the Japanese? Are there means – in addition to IRS – to support such a faster, technical exchange?

Answer 3: USNRC staff has interacted with the European Clearinghouse and attended their expanded EU Clearinghouse kick-off meeting in April, 2010. We exchange information with the EU Clearinghouse representatives at regular meetings of the OECD/NEA/CNRA Working Group on Operating Experience a group with representation from the U.S. and international nuclear regulatory authorities. This working group was formed by the Committee of Nuclear Regulatory Authorities (CNRA) within the Nuclear Energy Agency (NEA), which is part of the Organization for Economic Cooperation & Development (OECD), headquartered in Paris France. European Clearinghouse representatives also routinely attend meetings of the National Coordinators in the International Reporting System for Operating Experience (IRS) which is a web-based automated system for reporting and evaluating nuclear events and operating experience, a joint initiative of the NEA and the International Atomic Energy Agency (IAEA). The USNRC has provided staff expertise to help conduct learning workshops held by the EU Clearinghouse and has provided data to the EU Clearinghouse in the form of Operating Experience reported to the IRS. The EU Clearinghouse makes use of U.S. Licensee Event Reports and other publically available documents as data inputs to its studies. USNRC intends to maintain this current level of involvement and interaction with the EU Clearinghouse.

Question 4: Does the manufacturer have requirements to issue Engineering Change Notices when they change design and do they have to have a Correction Action Program to prevent recurrence? Thanks.

Answer 4: There are no direct and specific requirements set for the manufacturers in the Finnish legislation regarding Engineering Change Notices and Corrective Action Program. However, there are requirements for implementing a Management System that should include ECN and CAP processes. Requirements for MS has to be applied by all organizations which produce services or products that have safety significance (safety classified).

Question 5: What prompted the licensee to change the valves at OL 1&2? Was it part of a planned upgrade or a result of degraded performance?

Answer 5: Reason for modification is both degradation and upgrade. With a modified design licensee is able to change the valve bushings at site instead of shipping the valve to manufacturer.

Question 6: It is important to reflect OpEs appropriately to regulations; however, OpEs below a certain level should be handled by utilities or industry organizations such as INPO. Could you please let me know the threshold level that regulators screen in/out OpEs, and how the levels were developed in STUK?

Answer 6: Current regulatory requirements for event reporting are described in YVL 1.5 (http://www.edilex.fi/stuklex/en/lainsaadanto/saannosto/YVL1-5). In general, it has three event categories. Safety significant events (Emergencies, Events related to radiation safety, Situation related the Operating Limits and Conditions, Loss of safety function, Damage of safety significant SSCs, Weaknesses in safety management), Disturbance reports (Reactor scram, turbine scram, other event causing more than 5 % power decrease) and other incidents (near misses etc.). Special reports and Disturbance report shall be submitted to STUK. Other incident reports are not categorically required to be reported to STUK. STUK is currently updating the event reporting criteria.

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Session Day and Time: Tuesday, 1:30 pm–3:00 pm

Session Number and Title: T7 Status and Path Forward on Generic Safety Issue-191, "Assessment of Debris Accumulation on Pressurized-Water Reactor (PWR) Sump Performance," and Implications for Boiling-Water Reactors (BWRs)

Session Chair: William Ruland, Director, Division of Safety Systems, NRC/NRR

Session Coordinator: Stewart Bailey, NRC/NRR, tel: (301) 415-1321, e-mail: Stewart.Bailey@nrc.gov

Question 1: You stated that the Germans & Swedish utilities have closed the containment sumps strainer performance issue. How are the solutions that have been used different than the three options being considered for the US regulated utilities? Is their data available or lessons learned that we can utilize from our international counterparts.

Answer 1: In general, the German and Swedish solutions rely on backwashing or otherwise cleaning the strainer to remove accumulated debris, while the U.S. solutions do not. If a licensee proposed a similar approach, the NRC staff would review it on a plant-specific basis. Additional information on the German solution can be found under ADAMS Accession No ML101540090. Additional information on the Swedish solution can be found at http://www.oecd-nea.org/nsd/docs/2002/csni-r2002-6.pdf and the link at http://www.oecd-ilibrary.org/nuclear-energy/debris-impact-on-emergency-coolant-recirculation/results-of-tests-with-large-sacrificial-and-self-cleaning-strainers-and-the-installation-at-ringhals-2_9789264006676-9-en;jsessionid=7e6e3ijprre8o.delta.

Question 2: No information was provided with respect to the ZOI testing currently in progress. Can you provide an update summary?

Answer 2: The PWR owners group discussed their plans for zone of influence testing during a meeting with the NRC staff on March 3, 2011. Much of this meeting focused on a proprietary method for translating the jet testing into a zone of influence. More information can be found in the non-proprietary slides at ADAMS Accession Nos. ML110690437 and ML110690440.

Question 3: Do Risk Models Treat LBB-Qualified piping the same as NON-LBB Qualified Piping? At replicate units, would the risk numbers be identical if one had qualified RCS piping 12" and above, and the other had qualified RCS piping down to 6" and above?

Answer 3: Current PRAs use primary coolant system LOCA frequencies based solely on equivalent rupture diameter and reactor type (PWR versus BWR). Large break LOCAs are generally 6 inches and greater and use one frequency estimate which is much lower than frequency estimate for smaller break sizes. The expert elicitation report (NUREG-1829, found at ADAMS Accession Nos. ML080630013 and ML081060300) provides the latest LOCA frequency estimates based on equivalent diameter and reactor type. The elicitation process included consideration of many of the same type of large bore piping properties used in support of LBB qualification. The NRC currently has no basis for explicitly crediting LBB systems, but this could be proposed by a licensee.

Question 4: Is NRC doing research to determine whether plugging source term is conservative or overly conservative?

Answer 4: The NRC is not currently performing research to determine the level of conservatism in the source terms. In their RIC presentations, the PWROG and BWROG discussed industry research and testing that could be used to assess the degree of conservatism in the debris source terms.

Question 5: How do you intend to make progress for a better qualification of sump clogging probability?

Answer 5: The NRC staff is reviewing the work being performed by the South Texas Project related to a risk-informed evaluation of GSI-191. A presentation was given at the RIC. Additional information can be found in the licensee's presentation slides for a February 22, 2011, meeting with the NRC (ADAMS Accession No. ML110550395).

Question 6: If as little as 50g fiber per assembly can block the bottom of the core, is there any way to reduce the fiber to reach strainer and get into the core? Is it possible to reduce fiber amount reaching the strainer by placing removable debris catchers on the containment floor?

Answer 6: There are several methods of reducing the amount of debris that reaches the strainer and the core. Some licensees have removed or modified insulation, added curbs or debris interceptors, or modified flow paths inside containment. Some strainer designs have features that reduce the amount of fiber that could bypasses the strainer and reach the core. Removable debris catchers on the containment floor may be an effective method of reducing the debris that reaches the core.

Question 7: Fuel assembly (FA) tests will give a maximum fiber amount to avoid FA clogging. What about size distribution of fiber. Is size distribution important? If yes, what about the impact on sump strainer performance test already approved? How to demonstrate that the accepted strainers remain adequate?

Answer 7: The size distribution that reaches the core is an important parameter for in-vessel effects testing. The size distribution of debris used in strainer head loss testing is based on the debris from steam and air jet testing. The size distribution of debris used for in-vessel testing is based on the debris that bypasses the strainers. The debris for in-vessel testing is generally finer than the debris for strainer testing. In general, the finer debris distribution is more limiting.

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Session Day and Time: Tuesday, 3:30 pm–5:00 pm

Session Number and Title: T9 Current Topics in Probabilistic Risk Analysis

Session Chair: Gary DeMoss, Branch Chief, Division of Risk Analysis, NRC/RES

Session Coordinator: Keith Tetter, NRC/RES, tel: (301) 251-7605, e-mail: Keith.Tetter@nrc.gov

Question 1: Do you anticipate there being a way to learn from overseas operating experience with New Reactors with respect to PRA technical adequacy?

Answer 1: There are several requirements in the ASME/ANS PRA standard regarding the use of operating experience in the design-specific and plant-specific PRA. Generally, at the new reactor design stage, generic operating experience data have been deemed by the NRC staff to be acceptable. The use of operating experience from foreign yet similar reactor designs may be acceptable to augment the generic data used in the PRA. But foreign operating experience would not substitute for plant-specific data.

Question 2: Sometimes a PRA makes conservative approximations when low frequency is involved. To what extent should best estimate methods be used versus conservative methods in terms of PRA Quality?

Answer 2: In general, conservative assumptions can be found in any PRA, particularly when low frequency is involved. These conservatisms can occur due to a lack of data, lack of phenomenological understanding, or to simplify the PRA modeling. Whether a best estimate method should be used is dependent on the significance of the conservatism with regard to its impact on the PRA results and its affect on the application of the PRA. If the conservative assumption can have a significant impact on the results such that the associated changes in the PRA results can influence the decision under consideration, then it may be prudent to use a best estimate method. For example, if the assumption may change the risk profile such that the acceptance guidelines for the application are not met and if a best estimate method results in the acceptance guidelines being met, then it may be prudent to use a best estimate (with appropriate consideration of uncertainties). In this case, the licensee would typically seek to improve their PRA modeling to use a best estimate approach in this aspect of the PRA. However, when addressing change in risk metrics, sometimes conservatism in the base PRA model may not reflect the actual increases in risk associated with an application. Consequently, a conservative approach can result in a non-conservative change in the risk calculation that may indicate acceptance guidelines are met; however, when, if a best estimate model was used, the risk change would not meet the acceptance guidelines. In this case, the NRC staff typically evaluates the potential impact this conservatism in the base PRA model has on the decision being made. The staff may request additional information from the licensee that can include sensitivity calculations or other appropriate analyses and justifications from the licensee to support the application.

Question 3: How do you explain to the general public terms such as PRA, Root Cause Analysis, and defense in depth, etc.?

Answer 3: These terms can be confusing in that there are different interpretations used for these terms across the industry. The NRC is developing a Glossary with an objective of identifying and defining terms that are used in risk-informed activities related to commercial nuclear power plants (these terms are in the glossary). This glossary is intended to provide a single source where terms can be found. A major goal of the glossary is to reduce ambiguity in the definition of terms as much as possible, so that a common understanding can be achieved which will facilitate communication regarding risk-informed activities. Among other things, this glossary should allow individuals to distinguish communication issues, erroneously perceived as technical issues, from actual technical discussions. Where terms are found to have a justifiable variety of definitions depending on the context in which they are used, the glossary will also explain the individual definitions along with the context, to assure proper context-specific use of the term. Further, it is the intent of the glossary to provide a definition in "plain language;" that is, a definition is provided that does not rely on technical jargon. The reason the definition is written in plain language is to help ensure that there is no misunderstanding of the definitions. Furthermore, plain language helps PRA practitioners, including those who are not native English speakers and members of the public, to understand the definitions with minimum language barrier or PRA experience.

Question 4: Has EPRI looked at the March 2010 Robinson Fire event to see how well the EPRI HRA methods predicted the operator failures during this event?

Answer 4: EPRI has not yet performed an evaluation of the conditions associated with the March 2010 Robinson fire event using current HRA methods. It should be noted, however, that HRA methods do not typically "predict" operator failures. They do (or should) highlight situations that present particular challenges to an operating crew.

Question 5: As part of this new HRA effort, will EPRI allow external review of data that was used in the 1990's to develop the HCR/ORE failure rate/HEP data?

Answer 5: The data collected under the Operator Reliability Experiments and the use of these data in formulating the HCR/ORE correlation are reported in three volumes of EPRI NP-6937. This report is publicly available, and can be downloaded from the EPRI web site.

Question 6: How much of this Level 3 PRA science is available to the petroleum industry bearing in mind the recent accident in the Gulf of Mexico and the need to drill for oil in environmentally sensitive areas?

Answer 6: PRA is a structured, analytical process that *can be applied to any system* to provide both qualitative insights and a quantitative assessment of risk by: (1) identifying potential initiating event sequences that can challenge system operations and that can lead to an adverse event (e.g., onset of reactor core damage, radioactive material release to the environment, health and economic effects); (2) estimating the likelihood of these sequences; and (3) estimating the consequences associated with these sequences.

The PRA methodology that has been used by the NRC and the commercial nuclear power industry for the past three decades is well known and widely accessible to the general public. NUREG/CR-2300, "PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants" (January 1983), introduced the concept of dividing a PRA for a nuclear power plant into three sequential levels of analysis. This document, much of which is still relevant today, can be accessed on the NRC's Public Website using the following hyperlink: NUREG/CR-2300, "PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants".

Question 7: Given all the risk activities: NFPA805, 50.69, RI-TS5B and 4B, New RAs, etc.; that will be keeping the staff very busy for at least the next 5+ years, what resources are going to be used for the Level 3 PRA study and how many FTE are estimated to be needed?

Answer 7: The NRC staff is currently developing various options for proceeding with future Level 3 PRA activities. The resources needed to support each of these options, both in terms of contract dollars and staff full-time equivalents (FTE), will depend on many factors, including: objectives, scope, PRA technology to be used, site selection attributes, and staff capability.

By July 2011, the staff plans to provide the Commission with a SECY paper that will identify various options for proceeding with future Level 3 PRA activities, which will include resource estimates and perspectives on future uses for Level 3 PRAs. The staff plans to make this SECY paper available to the public.

Question 8: Some practitioners argue that there is a need for additional methods and data development prior to starting a full scope Level 3 PRA. For example, Seismic Risk Analysis, Human Reliability, etc. How does the NRC staff respond to this?

Answer 8: The NRC staff agrees that enhancements could be made to existing PRA methods, models, tools, and data to ensure future site Level 3 PRAs are of sufficient quality to support a wide variety of regulatory applications. Examples include: consequential (linked) multiple initiating event modeling; multi-unit modeling; post-core damage human reliability analysis

(HRA) modeling; and non-reactor PRA technology (e.g., success criteria determination, HRA, accident phenomenology, and source term analysis for spent fuel PRA).

However, these enhancements could be made either prior to or in parallel with commencing a site Level 3 PRA. The staff is considering both possibilities.

By July 2011, the staff plans to provide the Commission with a SECY paper that will identify various options for proceeding with future Level 3 PRA activities, which will include resource estimates and perspectives on future uses for Level 3 PRAs. The staff plans to make this SECY paper available to the public.

Question 9: Will the scope of the proposed PRA study include multiple accidents at multi-module SMR facilities?

Answer 9: The NRC staff is currently developing various options for proceeding with future Level 3 PRA activities. As part of this effort, the staff is considering options that might include an analysis of accidents involving multiple units at multi-unit sites, including multi-module small modular reactor (SMR) facilities.

By July 2011, the staff plans to provide the Commission with a SECY paper that will identify various options for proceeding with future Level 3 PRA activities, which will include resource estimates and perspectives on future uses for Level 3 PRAs. The staff plans to make this SECY paper available to the public.

Question 10: Data on latent cancer deaths have so much variability, how can we achieve any meaningful results from a Level 3 PRA? For example, if we conservatively use studies that a relatively small dose will kill in 20 years, how have we helped?

Answer 10: It is Commission policy to use mean value point estimates of various measures of risk, or "risk metrics," when using PRA results as part of the regulatory decision making process. It is also Commission policy that appropriate consideration be given to the uncertainty in the results.

PRA models cannot perfectly represent the real world; nor can they account for the unknown. Assumptions and approximations must be made to keep the models manageable. Uncertainty is therefore an inevitable component of any PRA. When using PRA results as part of any regulatory decision making process, it is therefore important to understand the types, sources, and potential impact of uncertainties associated with PRA models and how to treat them in the decision making process. NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making" (March 2009), was developed by the NRC to provide guidance on how to address these issues. The guidance provided in NUREG-

1855 is generic and independent of the source of uncertainty, and therefore would apply to a future Level 3 PRA.

Finally, quantitative estimates of various risk metrics are only one output of a PRA. PRAs can also yield qualitative insights that may actually be more important than the quantitative results. For example, in addition to providing a quantitative estimate of overall risk, a full-scope integrated site Level 3 PRA can provide valuable insights into the relative importance of all nuclear power plant site risk contributors to focus critical resources on those risk contributors most important to public health and safety.

For more information, the interested reader is referred to the following publicly available documents:

- Commission Policy Statement: "Safety Goals for the Operations of Nuclear Power Plants" (51 FR 30028)
- NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making"

Question 11: Could you please explain what weaknesses in prior PRA studies will be corrected by the New Site Level 3 PRA?

Answer 11: The last NRC-sponsored Level 3 PRAs were conducted in the late 1980's and documented in a collection of NUREG/CR reports and a single corresponding summary document, NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants" (December 1990). NUREG-1150 provides a set of PRA models and a snapshot-in-time (circa 1988) assessment of the severe accident risks associated with five commercial nuclear power plants of different reactor and containment designs. Still regarded by many as a significant contribution to advancing the state-of-the-art in PRA methods, models, tools, and data–collective referred to as "PRA technology"–the NUREG-1150 study significantly influenced and shaped the NRC's transition to the existing risk-informed regulatory framework.
The staff has identified several compelling reasons for proceeding with new Level 3 PRA activities that can be organized into three broad categories: (1) modifications to enhance nuclear power plant safety and security; (2) advances in PRA technology; and (3) additional scope considerations.

- ***Modifications to Enhance Nuclear Power Plant Safety and Security***
  PRA models should strive to be as realistic as practicable, representing the as-designed, as-built, and as-operated plant. Over the past two decades, the increased use of PRA results and insights by both the nuclear industry and the NRC has helped to improve NPP safety and operational flexibility. In addition to the implementation of multiple risk-informed regulations, there have also been a number of modifications to plant design, maintenance practices, operating and emergency procedures, severe accident management and extensive damage mitigation strategies, and training practices that

have enhanced both the safety and security of nuclear power plant sites. None of these substantial improvements were reflected in the NUREG-1150 PRA models.

- ***Advances in PRA Technology***
  Likewise, along with the acquisition of over 20 years of operating experience and insights gained from relevant severe accident and PRA-related research, there have been significant advances in PRA technology. These advances in knowledge and PRA technology should result in improved methods, models, tools, and data with an associated reduction in uncertainties.

- ***Additional Scope Considerations***
  Although they were considered to be full-scope at the time, the NUREG-1150 PRAs were limited in scope. These PRAs were limited to the assessment of single-unit reactor accidents initiated primarily by internal events occurring during full-power operations. A limited set of external events (fires and earthquakes) were considered for only two of the five analyzed nuclear power plants.

To update and improve its understanding of nuclear site accident risks, the NRC is considering evaluating accidents that might occur during any plant operating state, that are initiated by all possible internal events and external events, that may simultaneously affect multiple units per site, and that may affect multiple site radiological hazards (e.g., spent nuclear fuel).

Question 12: What is the NRC's position on living PRAs? Living PRA tools uploaded on a frequent basis (i.e. day to day) to reflect actual plant status and to be used for making operational decisions.

Answer 12: The NRC does not require licensees to maintain a PRA. However, when using PRA results as part of the regulatory decision making process, it is Commission policy that these results be derived from a PRA model that represents the as-designed, as-built, and as-operated and maintained plant to the extent needed to support the application.

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Session Day and Time: Tuesday, 3:30 pm–5:00 pm

Session Number and Title: T11 Low-Level Waste Management—Aligning Regulations with Current Reality

Session Chair: Larry Camper, Director, Division of Waste Management and Environmental Protection, NRC/FSME

Session Coordinator: James Kennedy, NRC/FSME, tel: (301) 415-6668, e-mail: James.Kennedy@nrc.gov

Question 1: Can we be speculative on when the Texas facility will be able to receive waste and what types of waste?

Answer 1: The facility operator, Waste Control Specialist LLC, has projected construction completion for the commercial Texas Compact waste disposal facility by the end of 2011. Prior to receiving waste, the license requires approved financial assurance to be in place, state inspection of the facility for compliance with all standards, submission of an updated performance assessment and other required reporting items. The current low-level radioactive waste disposal license issued for operation of the WCS site, does not address imported waste streams into the Texas Compact waste disposal facility, and only authorizes waste streams characterized and projected from Texas and Vermont waste generators, with additional license limitations. A license amendment would be necessary to allow for the acceptance and disposal of any imported waste. The type of license amendment needed would be dependent on the waste streams and the requested action being proposed associated with the importation consistent with TCEQ license amendment rules.

Question 2: What is TCEQ's position regarding blending of LLW and the potential of disposal of blended waste at WCS?

Answer 2: Texas has had a long-standing policy that radioactive waste diluted in any way in order to meet waste acceptance criteria for disposal, must retain its original classification if disposed within the state. The TCEQ has not taken a position on the blending rulemaking other than NRC recognition of current state policy and ongoing flexibility in any compatibility requirement for Agreement States regarding blending in order to maintain state policies that have allowed successful disposal facility siting and licensing.

Question 3: What would the view of Texas be if 61.58 was given the same level of compatibility as 61.55 giving the State the discretion to use the PA to override the classification tables?

Answer 3: Texas, as an Agreement State, has used the flexibility inherent in 10 CFR Part 61 for developing state rules and policies for disposal that have been acceptable to state policymakers and the public. Texas is currently using site-specific performance assessment in concert with the classification tables as the foundation of the issued license for low-level radioactive waste disposal. There is no need to have one regulatory tool override another when both can be used together to build public confidence and provide protection of public health and safety and of the environment.

Question 4: What barrier code would be used for the shotcrete evaluation? Time period? Background: The policymakers thought shotcrete would help contain LLW for public perception/acceptance. However, it may be counterintuitive based on evaluation.

Answer 4: The reinforced concrete barrier is designed to be shotcrete on the disposal unit floor (including the berms), side slopes, and as part of the cover system. The barrier is constructed of epoxy coated welded wire fabric and high strength shotcrete. The geostructural behavior of this barrier was demonstrated in the structural stability modeling using the FLAC code (Fast Lagrangian Analysis of Continua). This code analyzed the long-term performance of the disposal unit including interaction between the soil, shotcrete, and concrete canisters for the accumulated effect of concrete creep over 300 years and random patterns of canister failures, as well as seismic performance. The technical specification states that the design, construction, testing, and inspection of the shotcrete containment structure will comply with each of 24 listed codes and standards. The specification requires compliance with the standard ASTM A820: Standard Specification for Steel Fibers for Reinforced Concrete. Assuming adherence to the relevant codes and standards, the shotcrete barrier should have a 300-year design life. For purposes of performance assessment modeling, the shotcrete barrier is not relied upon beyond 300 years.

Question 5: To avert inadvertent intrusion, how long should institutional controls be? Background: At the engineered barrier performance workshop, August 3-5, 2010, Gary Robertson, WA DEP, suggested extending institutional controls from 100 years to comport with uranium mill tailings based on public hearings conducted in WA State.

Answer 5: The current Part 61 regulation allows for reliance on active institutional controls for a period of no more than 100 years following the closure of any LLW disposal facility. Agreement States may, in the implementation of Part 61, specify timeframes greater than 100 years but no credit may be taken for this extra time in demonstrating compliance with the Pat 61 performance objectives. As indicated in the question, it may be appropriate to allow credit for timeframes longer than 100 years. Stakeholders and other interested members of the public may choose to comment on this aspect of the current regulation when they submit comments and advice concerning the proposals outlined in SECY-10-0165.

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Session Day and Time: Tuesday, 3:30 pm–5:00 pm

Session Number and Title: T12 Regional Session—Operating Nuclear Power Plant Issues of Current Interest

Session Chair: Martin Virgilio, Deputy Executive Director for Reactor and Preparedness Programs, NRC/OEDO

Session Coordinator: Richard Barkley, NRC/RI, tel: (610) 337-5065, e-mail: Richard.Barkley@nrc.gov

Question 1: At one time there was an issue with SUNSI for license renewal applications. Has this been resolved?

Answer 1: The NRC conducts a SUNSI review of all license renewal applications received. Any issues that result from the review is addressed per established NRC procedures and regulations. There are currently no issues regarding SUNSI in license renewal applications at this time.

Question 2: Are there other areas where protection of SUNSI is an issue?

Answer 2: The NRC has established procedures and regulations that address the use and protection of SUNSI. Issues discovered at any point are resolved using these procedures and regulations.

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Session Day and Time: Tuesday, 3:30 pm–5:00 pm

Session Number and Title: T14 Strategic Considerations for Managing the Back End of the Fuel Cycle

Session Chair: Catherine Haney, Office Director, NRC/NMSS

Session Coordinator: Elizabeth Doolittle, NRC/NMSS, tel: (301), 492-3245, e-mail: Elizabeth.Doolittle@nrc.gov

Question 1: What is DOE doing to remove fuel from reactors to implement the NWPA? Do you know of any scientific reason to terminate Yucca Mountain?

Answer 1: The Department of Energy (DOE) is committed to meeting its responsibilities with respect to spent nuclear fuel and high-level waste. However, there is no evidence that continuing to pursue a repository at Yucca Mountain is the quickest or even a workable approach to meet these obligations. As explained in filings before the Nuclear Regulatory Commission and the Circuit Court of Appeals for the District of Columbia, the project suffered over many years from persistent opposition of the community. In light of this opposition and obstacles, there was considerable uncertainty about whether the Yucca repository could ever have been constructed and if so opened. Before NRC could have even considered whether to grant construction authorization for the repository, hundreds of contentions related to safety concerns would have had to have been adjudicated in contentions hearings. Even if a license were granted by NRC, before construction could commence, new legislation would have been required to withdraw the land for the repository and to appropriate money. In addition additional permits and water rights would have had to have had to be obtained. Even if the project overcame the continued community opposition and other hurdles and was constructed, it is uncertain whether the Department would ever have been able to obtain an operating license for the repository. Further contentious adjudicatory hearings would have been required before NRC could have considered granting such a license and the Department would have been required to obtain the necessary funding and permits and to build the 300 mile railroad necessary to transport the material to the repository.

The Department is committed to pursuing better, more workable alternatives to meet its responsibilities for the safe management and disposal of spent nuclear fuel and high-level waste. To that end the Secretary has established a Blue Ribbon Commission on America's Nuclear Future to conduct a comprehensive review of policies for managing the back end of the nuclear fuel cycle. The Commission will provide advice and make recommendations on issues including alternatives for the storage, processing, and disposal of spent nuclear fuel and high-level waste. Its interim report is due in the summer of 2011 and its final report in January 2012. Alternatives to a repository at Yucca Mountain may in fact take less time to implement and lead to shorter storage times at reactor sites.

The only way to open the path toward a successful nuclear future for the United States was to turn the page and look for a better solution - one that is not only scientifically sound but that also can achieve a greater level of public acceptance than would have been possible at Yucca Mountain. It is time to move beyond the 25 year old stalemate over Yucca Mountain - especially since technology has advanced significantly during that time, giving us better options both in terms of science and public acceptance.

Question 2: Is DOE involved with Korea in their reprocessing regime of taking LWR fuel and just crushing and resintering it into CANDU fuel?

Answer 2: No, the Fuel Cycle Research and Development program is not collaborating with the Republic of Korea on CANDU fuel.

Question 3: Won't legislation be required to change the National Policy that is set forth in the National Waste Policy Act?

Answer 3: DOE will assess the need to amend the Nuclear Waste Policy Act after receiving and reviewing the final recommendations of the Blue Ribbon Commission on America's Nuclear Future. Its interim report is due in the summer of 2011 and its final report in January 2012.

Question 4: In the systems engineering approach are risks associated with regulatory uncertainty captured? For example, the current lack of a cohesive regulation for reprocessing facilities makes it difficult to establish the economics of a reprocessing facility regardless of technology. So are economics included in any way in the systems model which considers regulatory risk?

Answer 4: The Office of Fuel Cycle Technologies is using the principles of systems engineering to develop a methodology that will be one of the tools used to make programmatic decisions on establishing research and development priorities. As this tool is matured it is intended to guide future decisions on down-selection of technologies to be demonstrated. We recently completed a pilot project, Initial Screening of Fuel Cycle Options, to develop and test the methodology. As part of this activity, we developed criteria to evaluate fuel cycle options according to their potential for meeting program objectives. Economics and "licensability" were part of the criteria. One of the conclusions of the pilot project is that economic criteria and associated metrics require further development and supporting analyses in order to realistically assess cost performance. As the insights gained from this pilot project are incorporated into the methodology to be used for future program decisions, the approach will capture the risks associated with regulatory uncertainty as well as with economic performance.

Question 5: You mentioned DOE's Used Fuel Disposition Program in passing. Please explain 1) how the UFD works and results relating to, or that will be used in the DOE fuel cycle research and development program; 2) how the rather loosely coupled tasks in the UFD Program will be better integrated in the future. Does it need BRC input?

Answer 5: Radioactive wastes generated by existing and future fuel cycles need to be safely stored, transported, and disposed. The Used Fuel Disposition (UFD) R&D program will identify options for performing these functions, including research into disposal in a variety of geologic environments. The R&D will consider used fuel and high-level waste inventories arising from the current reactor fleet and any additional new builds, including the potential for changing used fuel characteristics from enhanced operations (e.g., increased fuel burnup) and the projected inventories from advanced reactor and fuel cycle systems (e.g., High Temperature Reactors and Small Modular Reactors). This research is important to all of the potential fuel cycle approaches.

Sustainable fuel cycle options are those that improve uranium resource utilization, maximize energy generation, minimize waste generation, improve safety, and limit proliferation risk. The key challenge is to develop a suite of options that will enable future decision makers to make informed choices about how best to manage the used fuel from reactors. To this end, the Administration has established the Blue Ribbon Commission (BRC) on America's Nuclear Future to inform this waste-management decision-making process. Specifically, the Blue Ribbon Commission will provide advice, evaluate alternatives, and make recommendations for developing a safe, long-term solution to managing the Nation's used nuclear fuel and nuclear waste, and a new plan to address issues, including evaluating existing fuel cycle technologies and R & D programs, among other issues. The Blue Ribbon Commission will produce a draft report to the Secretary of Energy this summer and a final report with recommendations in January 2012.

Question 6: With the policy shown today, DOE does not aim at solving the spent fuel problem in due time, and will leave the legacy to future generations. Is there any consideration being given to minimizing the impact of nuclear energy on future generations?

Answer 6: The mission of the Fuel Cycle Research and Development program is to conduct research and development to help develop sustainable fuel cycles. Sustainable fuel cycles are those that improve uranium resource utilization, maximize energy generation, minimize waste generation, improve safety, and limit proliferation risk. We believe that the potential exists for making significant improvements in the cost effectiveness, proliferation risk, and environmental impact of current technologies. Since the U.S. Nuclear Regulatory Commission finds that used nuclear fuel can be stored safely for many decades to come, the Department of Energy is seeking significant improvements to the current technologies through its long-term research and development program.

Question 7: Openness and transparency are part of NRC's policy in communicating with the public. What are three ways that NRC is reaching out to stakeholders to communicate program changes?

Answer 7: NMSS intends to reach out to stakeholders by:

1) Providing information on NRC plans and key documents at conferences. Our recent RIC session and our presentation are an example of that kind of communication.
2) Ensuring that key documents that explain our plans are made publically available. Our recent plan for updating the waste confidence rule is an example of the kind of document that explains our plans and proposed directions.
3) Soliciting stakeholder input on specific activities by conducting meetings for public involvement. For example, we plan to have a stakeholder workshop in the summer of 2011 to solicit input on potential issues associated with extended storage and transportation of spent nuclear fuel.

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Session Day and Time: Wednesday, 1:30 pm–3:00 pm

Session Number and Title: W16 Construction Inspection Program

Session Chair: John Tappert, Deputy Director, Division of Construction Inspection and Operational Programs, NRC/NRO

Session Coordinator: Aida Rivera-Varona, NRC/NRO, tel: (301) 415-4001, e-mail: Aida.Rivera-Varona@nrc.gov

Question 1: Can you share anything relative to the public disclosure that an inspection of the facility in Louisiana was halted?

Answer 1: The NRC conducted a vendor inspection on January 10 through January 12, 2011, at the Shaw Modular Solutions (SMS) facility in Lake Charles, LA. The NRC inspection was terminated early due to the current status of activities at SMS. Public information on the NRC inspection at SMS can be found at http://www.nrc.gov/reactors/new-reactors/oversight/quality-assurance/vendor-insp/insp-reports/2011/.

Question 2: Is the NRC working with the international regulatory agencies to extend the approval of calibration and testing laboratories that are certified under ILAC agreement by authorized bodies? Is this being address through a multi-national agreement?

Answer 2: The NRC is actively reviewing implementation strategies to consider expanding NRC's recognition (beyond domestic accreditation bodies) to international accreditation bodies on the basis that they are all full signatories to the ILAC Mutual Recognition Agreement (MRA). In addition, the NRC is actively reviewing implementation strategies to include testing laboratories accredited under the requirements of International Standard Organization (ISO)/International Electrotechnical Commission (IEC) 17025, “General Requirements for the Competence of Testing and Calibration Laboratories,” as part of our recognition of the ILAC MRA process. At this time the process is not being addressed through a multi-national agreement.

Question 3: When is the next vendor workshop?

Answer 3: While the NRC plans to continue this series of workshops on vendor oversight for new reactor construction, we have not yet finalized the timing of the next workshop, but it will likely be in 2012.

Question 4: Regarding NRC overseas inspections, will NRC created inspection reports and documented inspection observations be publicly available?

Answer 4: NRC overseas inspection reports are available and can be found in the same location as domestic inspection reports. They are currently available at http://www.nrc.gov/reactors/new-reactors/oversight/quality-assurance/vendor-insp/insp-reports.html.

Question 5: Explain target ITAAC. Target ITAAC for AP1000 was based on Rev 15. Now that Rev 18 is being reviewed by NRC, are targeted ITAAC being re-evaluated? When will this occur?

Answer 5: Targeted ITAAC are determined using a prioritization process that evaluates the value of inspection for each ITAAC. There will be a re-evaluation of the targeted ITAAC for AP1000, although we will focus on the expected Rev 19 of AP1000. Only the ITAAC that have been changed or added from Rev15 to the expected Rev19 submittal will be newly prioritized.

Targeted ITAAC that were not edited or removed between these two revisions will remain as targeted ITAAC. The prioritization for Rev 19 is expected this spring after receipt of Rev 19.

Question 6: Is there a single database or process where ITAAC closure letters will be tracked?

Answer 6: Staff is planning to develop an internal database "VOICES" to track closure letters, supplemental closure letters, and 225-day uncompleted ITAAC letters. This will assist staff in tracking the complex task of managing all the ITAAC of a combined license. As a service to industry and the public, staff is also planning a website dedicated to posting the status of these submitted letters to the NRC. The website will update interested parties on the acceptability and closure status of each ITAAC of the combined license.

Question 7: Regarding hearing phase during the ITAAC process, are there any potential for delays? What preparations have been made to assure that hearing are thorough yet quick?

Answer 7: As mentioned in response to the question above, staff is planning to develop a website to keep the public informed of ITAAC completion status, minimizing the time by which the public is informed. Staff is also investigating different methods to document its determination on ITAAC, so that there will be records on each determination that an ITAAC has been, or has not been, successfully completed. The records may serve as supporting documentation for potential hearings, which would reduce the hearing preparation time. The use of Federal Register Notices, as mandated by 10 CFR Pt. 52, will also keep the public informed.

Question 8: Do the NRC construction inspection program activities also apply to non-reactor projects? Fuel Enrichment, MOX, etc.

Answer 8: Yes, the NRC construction inspection program applies to nonreactor projects. However, different activities may use different terminology and the regulatory requirements may differ in some areas.

Question 9: Is the construction inspection group limiting their inspection to safety related SSCs, or is non-safety related inspected as well?

Answer 9: The construction inspection program (CIP) is focused on inspection of safety related systems or components (SSCs) as they related to ITAAC (IMC 2503) and programmatic inspections (IMC2504). The programmatic inspections will cross-cut many construction & operational activities and will include programs that apply to both safety and non-safety related systems. Although lessons learned from past construction suggest that the NRC should primarily focus the inspection resources on safety related activities.

Question 10: To what extent does the construction inspection program cover commissioning oversight? Both prior and after fuel loading.

Answer 10: The construction inspection program will provide oversight prior to the Commission's 10 CFR 103.g finding that ITAAC have been met by the licensee. At that point construction is considered complete. Following the 103.g finding, the NRC will implement the reactor oversight process (ROP) for startup testing and plant operation.

Question 11: With the changing security environment, how are these aspects being handled? Does NSIR play a role in developing requirements and inspection guidance? Are there specific issues that you are currently addressing?

Answer 11: The NRC routinely reviews all of its processes and procedures to ensure the most up to date regulatory requirements are incorporated into staff guidance. Currently, NSIR is developing security operational program inspection procedures to incorporate into the Part 52 inspection program requirements.

Question 12: 3 months assignments in China/Finland/France are obligatory part of 2 yr inspection training program? Are such assignments based on bilateral agreements with STUK/ASN or with their licensees?

Answer 12: International assignments are not part of the formal qualification process for NRC construction inspectors. The international assignments are used to gain insights and experience on the construction of new reactor designs. The agreements are coordinated by the NRC Office of International Programs with the regulatory body of the host country.

Question 13: How is the communication between the Region and Headquarter during this process?

Answer 13: The Division of Construction Inspection and Operations Programs (DCIP) has established a number of weekly meetings and teleconferences with the Center for Construction Inspection (CCI) at Region II. We are always striving to find innovative methods to enhance communication such as using video conferencing, web based conference call support, and increasing the actual face to face time spent between head quarters and regional colleagues.

Question 14: Can final inspection and tests be performed in a module manufacturing facility?

Answer 14: The licensee is responsible for the closure of inspections, tests, analysis, and acceptance criteria (ITAAC). Many ITAAC require verification of "as-built" SSCs. However, some of these ITAAC will involve measurements and/or testing that can only be conducted at the vendor site due to the configuration of equipment or modules or the nature of the test (e.g., measurements of reactor vessel internals). For these specific items where access to the component for inspection or test is impractical after installation in the plant, the ITAAC completion documentation (e.g., test or inspection record) will be generated at the vendor site and provided to the licensee. Onsite activities for these ITAAC will likely be limited to receipt and placement of the component/module in its final location. Closure letters for such ITAAC would not be submitted to the NRC until after the component/module is installed in its final location. A closure letter relying on a record review of the inspections or tests at the vendor site should reflect consideration of issues documented during subsequent fabrication, handling, installation, and testing. A licensee intending to rely upon a vendor inspection or test to satisfy an ITAAC requirement must take care that such reliance is consistent with the applicable DCD, including the DCD definitions of relevant terms, such as “inspection,” “test,” and “as-built.” The licensee will provide schedule information to the NRC, including plans to perform certain ITAAC activities in vendor shops, so the staff can plan their inspection and ITAAC verification resources accordingly.

Question 15: What do you consider different about a module facility? Why isn’t it just like a component vendor inspection?

Answer 15: A modular facility is a vendor facility. Modular facilities were used during past construction of nuclear plants; the only difference now is the scale of the modules is expected to be much larger and more complex. The NRC plans to perform both vendor and region based ITAAC inspection at the modular facilities. The type and scope of each inspection will be based on the work being performed at the facility. All inspection results will be used to verify the licensee’s closure of ITAAC and implementation of their operational programs.

Question 16: Can you discuss the impact of modular construction on the inspection program?

Answer 16: Large scale modular construction requires some modification to the inspection scheduling and planning but is not anticipated to require any changes to the inspection program or subsequent procedures. The NRC plans to inspect ITAAC related SSCs regardless of their location.

Question 17: How will NRO engage with SMRs licensed under Part 50.

Answer 17: NRO is currently evaluating the required modifications that might be needed to the inspection program to accommodate small or modular reactors that may be licensed under Part 50. As with all NRC processes, the NRC will inform all stakeholders about proposed changes in the inspection program through outreach activities such as public meetings.

Question 18: How are design deviations during construction dispositioned aside from ITAAC?

Answer 18: Currently, the NRC is drafting guidance about the process for design changes for all activity during construction, and expects to have guidance published in the near future. The guidance will include licensee review of design deviations under 10 CFR Part 50.92 and appropriate additions to NEI Guidance document 97-01. The guidance will also include a process for licensees to submit a license amendment preliminary acceptance review. If the preliminary review is accepted by the NRC, then the licensee will be able to proceed with the design deviation at risk while the NRC completes a full review of the license amendment request. The NRC is also evaluating the whether the Enforcement Manual should provide for enforcement discretion during new plant construction.

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Session Day and Time: Wednesday, 1:30 pm–3:00 pm

Session Number and Title: W20 Risk-Informed Technical Specification Initiatives

Session Chairs: Donnie Harrison, Branch Chief, Division of Risk Assessment, NRC/NRR and Robert Elliot, Branch Chief, Division of Inspection and Regional Support, NRC/NRR

Session Coordinator: Jigar Patel, NRC/NRR, tel: (301) 415-3109, e-mail: Jigar.Patel@nrc.gov

Question 1: How does 50.59 apply to changing frequencies that have been relocated.

Answer 1: This question is related to risk-informed technical specification initiative 5B. The applicability of 10 CFR 50.59 should be the same as for other relocated requirements, (i.e., determine if 10 CFR 50.59 applies to the requirement or not). Since surveillance frequencies are not design/licensing basis features, 10 CFR 50.59 would not apply to them.

Question 2: What's going to happen with NUREG-1860?

Answer 2: The purpose of NUREG-1860 was to establish the feasibility of developing a risk-informed and performance-based regulatory structure for the licensing of future nuclear power plants (NPPs). As such, the NUREG documents a "Framework" that provides an approach, scope and criteria that could be used to develop a set of requirements that could serve as an alternative to 10 CFR 50 for licensing future NPPs. Follow-on actions have not been identified, however, as indicated during the plenary sessions, Commission Apostolakis is forming a task force to evaluate the risk-informed framework within the current regulations. This task force is expected to complete its work within the next year.

Question 3: Have I understood correctly that risk-informed can only result in prescriptive values to be relaxed and if risk results turn out to be greater than thought, values remain unchanged?

Answer 3: The comment is correct in that if the risk results are greater than established acceptance criteria for the prescriptive (original deterministic) condition, then the surveillance frequencies (for RITS 5B) and completion time (front stop for RITS 4B) do not have to be reduced. The initial conditions established in the licensee's Technical Specifications (i.e., the front stop on RITS 4B and the initial relocated surveillance frequency in RITS 5B) have been established and tested through many years of operational experience. Accordingly, they are considered acceptable minimum conditions. If the licensee determines it to be prudent, these initial conditions may be reduced to reduce the risk consistent with the RITS 4B or 5B processes. However, there is no requirement to make this adjustment.

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Session Day and Time: Wednesday, 1:30 pm–3:00 pm

Session Number and Title: W21 Safety Culture—Implementation of the Policy Statement, Perspectives on Recent Events, and Activities in Japan

Session Chair: Andrew Campbell, Deputy Office Director, NRC/OE

Session Coordinators:
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Question 1: Do you think whistle-blowing should belong to safety culture?

Answer 1: The Commission recently approved the staff's proposed final Safety Culture Policy Statement, which communicates the Commission's policy on safety culture to the agency's regulated community. In the Policy Statement, one of the traits that describes a positive safety culture is "Environment for Raising Concerns – a safety conscious work environment (SCWE) is maintained where personnel feel free to raise safety concerns without fear of retaliation, intimidation, harassment, or discrimination." For its internal safety culture, the NRC uses a similar concept called Open Collaborative Work Environment (OCWE). OCWE is described as an environment that encourages all employees and contractors to promptly speak up and share concerns and differing views without fear of negative consequences. Both a SCWE, in the case of NRC's regulated community, and OCWE, for the NRC itself, are very important elements of a positive safety culture, and the agency supports employees who engage in behaviors consistent with such environments, which may include whistleblowing.

Question 2: Are the NRC internal safety culture case studies available publicly? If so, how can they be accessed?

Answer 2: NRC safety culture staff is currently starting an initiative to develop case study types of tools and products, as a way to identify applicable lessons learned insights from recent events and incidents, including from other industries. Once this initiative is further along in the development process, the staff will be planning to put together products that can be made publicly available. How they will be distributed publicly is still to be determined, but most likely they will be posted on the NRC's safety culture public website (http://www.nrc.gov/about-nrc/regulatory/enforcement/safety-culture.html) as one of the means. Information and updates on about NRC's safety culture activities are regularly provided on this page.

Question 3: What aspect or provision of the Safety Culture Policy addresses the technical competence of personnel in the NRC or the industry?

Answer 3: The Commission recently approved the staff's proposed final Safety Culture Policy Statement, which communicates the Communication's policy on safety culture to the agency's regulated community. The Policy Statement contains a list of traits that are present in a positive safety culture. It's important to note that the traits are high-level description of the areas important to a positive safety culture, and there could be much more detailed information to describe each trait that would be applicable to different types of environments in the regulated community. Concepts related to technical competence may be applicable under several of the traits depending on the type of organization, for example continuous learning and personal accountability. In addition, the Policy Statement states that there may be traits not included that are also important in a positive safety culture, depending on the organization.

Because the Policy Statement is focused on the regulated community, the NRC itself is not directly covered in its scope. However, as discussed during this RIC session, NRC has taken a range of activities for continuously improving and strengthening its own safety culture. The NRC has a number of training, developmental, and knowledge management activities and initiatives throughout the agency to develop and enhance the technical competence of its staff. Examples of such activities include formal training, development, and qualification plans; training courses; on-the-job training opportunities; seminars; and many other professional development activities. The agency places strong value and emphasis on training and continuous learning, and encourages all staff to identify and engage in such activities and opportunities.

Question 4: What do you believe will be the single most significant benefit to safety that the safety culture policy statement will produce?

Answer 4: For some of the members of our regulated community (e.g., certain material licensees), the Safety Culture Policy (SCPS) Statement is the first time they are considering the concepts of safety culture. One of the most important benefits we will receive from this SCPS is a heightened awareness and an increased understanding of this very important concept. As the regulated communities apply the safety culture traits in the SCPS in their organizations, it is anticipated to result in enhanced safety performance.

Question 5: Sometimes we hear the terms, "positive" safety culture or "strong" safety culture. Are the adjectives redundant? Doesn't a site either "have" a safety culture or "not have" one?

Answer 5: Every organization has an organizational culture which consists of the acceptable values and behaviors of that organization. Every organization also has a safety culture. It may not, however, be a positive safety culture. With a focus on nuclear power plants, for example, if there are numerous operator workarounds, an attitude of "we'll do the minimum needed to keep a piece of equipment running", and a low number of employees who use a corrective action tracking system, this would be considered a weak safety culture or a safety culture in need of improvement. It's still a safety culture—just not the type of robust positive safety culture that can have an effect of lowering potential events from occurring.

Question 6: Please comment on the merits of developing a consensus standard for safety culture.

Answer 6: By having 16 external stakeholders involved in the 3-day workshop to come to a consensus for a definition and traits to form the basis of the SCPS, there was "buy-in" by all 16 stakeholders. When an individual or an organization is involved in the decision-making process, this instills an attitude of ownership that will help to make the proposed change a reality. There is less resistance to an idea or a change when an individual, or an organization, is involved in what that idea or change may be.

Question 7: Assuming industry and NRC can agree that a common language is an important objective to achieve, what are the next steps to achieve this common terminology and associated definitions around NSC (nuclear safety culture)?

Answer 7: NRC does agree that a common language is an important objective. The SCPS is a vehicle to be used by our regulated communities to begin to come to that common language between and among their various organizations. We have one initiative, for that purpose, which will begin shortly. NRC staff will be working with INPO and NEI to develop common language, based on the SCPS, particularly for use within the reactor power industry's documents as well as the NRC ROP documents.

Question 8: Will the NRC's safety culture policy or some version of it ever become a rule? If not, why not?

Answer 8: At this point in time, the NRC believes that a policy statement is the appropriate vehicle to express the Commission's expectations and begin the discussion of safety culture within our regulated communities. As the regulator, we do have the option of choosing to go further and consider rulemaking if that is appropriate at some future date.

Question 9: Are there specific plans concerning convergence of terminology with IAEA?

Answer 9: We are actively engaged with IAEA and have provided a presentation related to the SCPS at a recent IAEA Technical Meeting on Safety Culture. Although we do not have specific plans (nor does IAEA) to move this SCPS language into the documents IAEA produces, we do have plans to continue to be involved in the safety culture arena with IAEA and share our experience in this area.

Question 10: What advice/programs/documents would you offer to countries who are seeking to build civilian nuclear power programs?

Answer 10: With respect to nuclear safety culture, we are involved with international organizations such as NEA's working groups and IAEA's meetings/working groups for the purpose of sharing information related to safety culture. Countries who are engaged in new construction attend those meetings. The NRC's Office of New Reactors and the Center for Construction in Region II conduct their regulatory responsibilities guided by manual chapters and inspection procedures. Safety culture is directly looked at as part of the assessment of construction activities, as described in Inspection Manual Chapter (IMC) 2505 and IMC 0613. IMC 2506 "Construction Reactor Oversight Process General Guidance and Basis Document" is the basis document and it serves to gain a general perspective on how the construction inspection program works and how the different areas of construction-inspection are integrated.

Question 11: If the policy remains unenforceable and an organization is requested to perform a safety culture assessment which identifies problems; what would be the NRC's recourse if no improvement in safety is made at that facility?

Answer 11: Inspection Manual Chapter (IMC) 0305 contains guidance for addressing Action Matrix movement and holding open inspection findings. When risk significant Significance Determination Process (SDP) findings or performance indicator (PI) thresholds are crossed a supplemental inspection (95001, 95002, or 95003) will be performed in accordance with IMC 0305 guidance. IMC 0305 allows findings to be held open past four quarters in the Action Matrix when:

- "The corresponding supplemental inspection reveals substantive inadequacies in the licensee's (1) evaluation of the root causes of the original PI or inspection finding, (2) determination of the extent of the performance problems, or (3) actions taken or planned to correct the issue, then additional agency action, including additional enforcement actions or an expansion of the supplemental inspection procedure may be needed to independently acquire the necessary information to satisfy the inspection requirements."

If the problem is related to a substantive cross-cutting issue (SCCI), then the SCCI will remain open and the NRC will request the licensee to respond to the SCCI either in annual public meetings, or in writing. If beyond the third consecutive assessment cycle there is no indication that the licensee has corrected the deficiencies leading to the SCCI, then IMC 0305 provides the following guidance:

- "If an SCCI with the same CCA is identified beyond the third consecutive assessment letter, and all of the options proposed above have been exhausted, the regional office may consider additional actions (those not covered by the Action Matrix) to address the issue."

Question 12: How does safety culture address the unknown unknowns, or the "Black Swan" event?

Answer 12: It is difficult, if not impossible, to address an unknown, unknown; however, one goal of the policy statement is to raise awareness on issues that may have an impact on safety and encourage regulated entities to consider various scenarios that emphasize the safe and secure use of regulated materials as they develop their processes and procedures.

Question 13: Has any thought been given to developing a change management template for NEI 09-07? Such a template would be very helpful.

Answer 13: Yes. At our four regional workshops on implementing NEI 09-07, we will be discussing the experience of the four pilot plants in implementing NEI 09-07. The plans and processes they developed will be discussed and shared with workshop participants.

Question 14: What nexus do you see between recent events, such as H.B. Robinson, and:

a) Safety culture within industry?
b) The NRC's ability to make timely assessments of safety culture?
c) Why didn't the Reactor Oversight Process (ROP) identify precursors?

Answer 14:

a) To date, operating experience has demonstrated that there is a nexus between safety culture and events at NRC-regulated facilities. Because weaknesses in an organization's safety culture may contribute to an increased likelihood of having problems and more severe consequences when problems do occur, the NRC has a responsibility to consider safety culture as part of its oversight authority.
b) NRC does not make assessments of safety culture; licensee's are responsible for safety culture and are responsible for performing independent safety culture assessments if requested.
c) The ROP involves applying risk-informed regulation rather than solely deterministic regulation and using indicative measures of performance rather than predictive measures of performance. The ROP was developed to provide tools for inspecting and assessing licensee performance in a more risk-informed, objective, predictable, and understandable manner compared to the previous oversight process. It was intended to

provide adequate margin in the assessment of licensee performance so that appropriate licensee and NRC actions are taken before unacceptable performance occurs. Therefore, the ROP distinguishes licensee performance based on objective thresholds for regulatory response rather than providing predictive indications of licensee performance.

Question 15: Where are the basic ROP principles as risk-informed, objective, … incorporated in the current regulatory approach to safety culture evaluation?

Answer 15: The current ROP principles of good regulation which include being transparent, objective, risk-informed and performance-based were considered during development of the safety culture components and aspects in the cross-cutting areas. Each of the 9 components and accompanying aspects were evaluated with respect to each of the ROP principles to ensure that they aligned.

Determinations regarding safety culture by their intrinsic nature warrant some degree of subjective judgment. By contrast, the ROP was designed to be an objective, risk-informed performance assessment process. Within this context, the ROP principles are incorporated into the oversight of cross-cutting areas and the approach to safety culture oversight in the following ways:

(1) The NRC's approach to cross-cutting issues and safety culture oversight provides indications of performance in cross-cutting areas within the framework of the ROP. As such, those indications are developed and characterized when more-than-minor performance deficiencies are identified in accordance with the fundamental regulatory principles of the ROP.

(2) The NRC's approach to cross-cutting issues and safety culture oversight ensures that findings with safety culture aspects are developed and characterized in a manner that is as objective as is practicable within the ROP.

(3) The issues are of more than minor significance <u>and</u> they are risk-informed if they are documented in an NRC inspection report. Safety culture assessments are either requested or performed for some supplemental inspections associated with safety-significant performance issues and/or long-standing substantive cross-cutting issues.

(4) The issues are specifically outlined in an effort to be transparent to internal and external stakeholders.

Question 16: What would be the regulatory basis, within either 10 CFR or Atomic Energy Act, for regulating safety culture?

Answer 16: The NRC does not regulate safety culture. However, there is a clear nexus between safety culture and poor plant performance. This initially resulted in the Commission directing the staff to enhance the ROP to more fully address safety culture. Currently, inspection findings are assigned associated cross-cutting aspects.

Question 17: If safety culture is not regulatory-based, not enforceable, how can negative cross-cutting findings result in greater levels of license response; perhaps greater scrutiny, and taking remedial actions, be required under the ROP?

Answer 17: The cross-cutting aspects of findings do not equal safety culture, although the NRC can glean insights into safety culture from them. Though not regulatory-based or enforceable, aspects of licensee performance such as human performance, the establishment of a safety conscious work environment (SCWE), and the effectiveness of licensee problem identification and resolution programs are important to meeting the agency safety mission. These items generally manifest themselves as the root causes of performance problems. The purpose of identifying a substantive cross-cutting issue (SCCI) in one of the three cross-cutting areas is to inform the licensee that the NRC has a significant level of concern with the licensee's performance in the cross-cutting area. The intent is to identify the issue for licensee attention so that appropriate actions can be implemented before degradation in plant performance results in an escalation of the agency's regulatory response. Although SCCIs would not result in enforcement action, the ROP provides for inspection and follow-up of licensees' responses to SCCIs.

Additionally, the ROP provides tools for assessing and responding to licensee performance in a manner that is more risk-informed, objective, predictable, and understandable than previous oversight processes, which were more compliance-based. The ROP provides oversight of performance issues and not just compliance issues. Several aspects of the ROP including cross-cutting aspects, reflect this concept, *e.g.*, documenting and assessing the significance of inspection findings associated with licensees' failures to meet self-imposed standards rather than solely failures to meet regulations and assessing performance indicators, which are voluntarily reported by the industry and not necessarily associated with regulations. These are just a couple examples of ROP tools, along with SCCIs that are not solely compliance based.

Question 18: Do you expect licensees to apply the safety culture components and assessments to security organizations at reactor sites?

Answer 18: The ROP currently identifies cross-cutting aspects that are associated with security findings and when a safety culture assessment is requested, it is expected that the assessment will include the security organization.

Question 19: One objective of the ROP was to focus stakeholders on issues of increased significance as determined by the SDP. To that end, the industry was assured that "green" findings (very low safety significance) would not be aggregated. However, the current safety culture process aggregates CCA's associated with findings (including green). This has had the effect of drawing increased attention to green findings (where most CCA's reside) and thus redirecting focus on greater then green findings. What, if anything, is being explored to restore stakeholder focus on greater then green findings? (E.g., Is there any consideration for CCA "weighting" based on the color of underlying findings?)

Answer 19: Although a "weighting" factor is currently not being considered, the staff conducts a yearly ROP self assessment and could consider it in the future. The focus of an SCCI is not the green findings associated with the cross-cutting aspects, but rather the cross-cutting theme and the lack of confidence the staff has in the licensee's scope and progress in addressing the cross-cutting theme. NRC regulatory actions per the Action Matrix are not taken in response to these SCCIs alone; however, they can influence the range of actions taken after PI and inspection thresholds are crossed. This influence, for example, can be in the form of adjusting the scope of the supplemental inspection performed in response to white inspection finding to focus some inspection effort on the performance deficiencies highlighted by a previously documented significant adverse trend in a cross-cutting issue. (Note, insight into this question can also be gained from the answer to question #4).

Question 20: How much are taxpayers paying for all this internal safety culture stuff?

Answer 20: The NRC is statutorily required to recover most of the agency's budget authority through fees assessed to applicants for an NRC license and to holders of NRC licenses. The Omnibus Budget Reconciliation Act, as amended, requires that the agency recover approximately 90 percent of our budget authority through fees, less monies appropriated from the Nuclear Waste Fund. Hence, the NRC's internal safety culture activities are supported through the agency's licensing fees. See additional information at: http://www.nrc.gov/about-nrc/regulatory/licensing/fees.html.

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Session Day and Time: Wednesday, 3:30 pm–5:00 pm

Session Number and Title: W23 Cyber Security—Licensing and Implementation at Nuclear Power Plants

Session Chair: Craig Erlanger, Branch Chief, Division of Security Policy, NRC/NSIR

Session Coordinator: John VandenBerghe, NRC/NSIR, tel: (301) 415-7142, e-mail: John.Vandenberghe@nrc.gov

Question 1: Can you address the EMP vulnerabilities and what is done/ or to be done about it?

Answer 1: The NRC recently updated an assessment of the electromagnetic (EM) vulnerabilities of nuclear power plant digital safety systems. This assessment indicates that the digital safety systems would not be adversely affected to the level where they could not perform their safety function.

Quality design processes to protect nuclear plant digital safety systems from electromagnetic interference and general construction techniques (such as required for potential seismic events) are expected to provide an adequate level of protection to the digital safety systems from fields that would be generated by an EM event.

Question 2: How are the licensees addressing the vulnerabilities uncovered by the NRC in digital I&C systems?

Answer 2: Licensees are implementing a Cyber Security Program in accordance with 10 CFR 73.54 requirements. All vulnerabilities discussed during the presentation on Digital Platform Cyber Vulnerability Research can be mitigated or eliminated by applying the security controls included in Regulatory Guide 5.71. The research was conducted using partial digital platform mockups, not an actual licensee installation. Therefore, different plant specific installations could require different measures which licensees will determine as part of their cyber security program efforts. The NRC will also inspect cyber security.

Question 3: What is being studied to protect digital system from the threat of a high altitude electromagnetic pulse (HEMP) detonation?

Answer 3: The NRC's recent EM vulnerability assessment considered such EM events as HEMP, other man-made EM events, and geomagnetic events. This assessment indicates that the digital safety systems would not be adversely affected to the level where they could not perform their safety function. Additional studies are not being performed at this time.

Question 4: How is defense-in-depth applied to cyber systems? Please elaborate.

Answer 4: In principle, defense-in-depth (DiD) suggests that the failure of any one system or protective measure does not compromise the entire system or put at risk what you are trying to protect. For cyber this means that physical access is one layer while a firewall is another. Two additional examples would be the use of strong passwords and anti-virus software. See Section C.3.2 and Appendix C, page C-12 of RG 5.71, "Cyber Security Programs for Nuclear Facilities." This document is publically available in ADAMS under ML090340159.

Question 5: Part 73.54: Does it include biometrics and digital fingerprinting?

Answer 5: 10 CFR Part 73.54 does not include "biometrics and digital fingerprinting." However, 10 CFR Part 73.57, in part, requires that NRC licensees authorized to operate a nuclear power plant shall fingerprint each individual who is permitted unescorted access to a nuclear power facility.

Question 6: Is there a published list of "standard" threat vectors? If so, where?

Answer 6: Threat vectors can be deduced given the cyber security controls promoted by standards organizations, such as the National Institute of Standards and Technology (NIST). Threat vectors are specific to the design of an enterprise combined with the way an organization functions and the trust models used. Therefore, each organization's vectors will likely vary.

Question 7: We constantly hear about "defense-in-depth" for cyber security, which is a good and correct. Are there any active offensive plans or ideas to have techniques to locate a hacker from outside and counterattack or lead to identification (of the attacker) or disabling the attack?

Answer 7: Finding any of the digital clues left behind by a would-be attacker requires good forensics. Unfortunately, the information to determine the actual source of an attack is not always available, or possible. There are many methods attackers can use to prevent the accurate identification of the actual source. Additionally, many systems that are used to attack other systems are actually compromised computers belonging to an innocent third party. Active attempts to compromise those machines could result in significant adverse effects on the innocent party.

Therefore, when it comes to launching an offensive strike, determining the actual source of the attack (attribution) is still a problem and the probability of causing harm to innocent third parties is very high. However, if certain organizations are provided with adequate forensics data, this will represent the best chance of preventing the attack in the long-term.

Question 8: Given the disparity in terms of digital control systems between the current fleet and new reactor designs, what is the qualitative assessment of overall potential cyber vulnerabilities for the current vs. new reactors? Is the level of effort being expended commensurate with the risk?

Answer 8: 10 CFR 73.54 and its implementing guidance, Regulatory Guide 5.71, provide a performance-based approach to compliance that allows both operating reactors and new reactors (with much more digital systems) to develop and implement a cyber security program tailored to the site and commensurate to the risk in all cases. The results of applying 10 CFR 73.54 to a new reactor versus an operating reactor may look very different, but in both cases a high assurance of protection against a cyber attack will be achieved.

Question 9: Since many of the cyber systems will come from overseas, what is being done to ensure supply chain risks are minimized?

Answer 9: In Appendix C to Regulatory Guide 5.71, Section 12 addresses supply chain issues, including the following:

- system and services acquisition policy and procedures,
- supply chain protection,
- trustworthiness,
- integration of security capabilities,
- developer security testing, and
- licensee/applicant testing.

Question 10: Has the NRC considered using White Hat aggressor teams in its inspection program?

Answer 10: Currently the NRC is in the process of developing its inspection program. Any process and protocol which enables the NRC verify the licensee is protecting Safety, Important to Safety, Security and Emergency Preparedness functions are being considered.

Question 11: What kind of significance thresholds do you envision for the SDP? Would they be risk-informed consider core damage frequency (CDF)/large early release frequency (LERF) from a successful attack on one or more targets?

Answer 11: The NRC will begin developing in the latter half of 2011, the cyber security significance determination process. The process will be based on the performance objective of the licensee's cyber security program which is to provide high assurance that the licensee can protect against both external and internal cyber attacks. Licensee performance in the security cornerstone is assessed by determining the significance of security-related issues and findings relative to the performance objective.

Question 12: Have there been any cyber attacks at a Nuclear Power Plant? If so, what happened? Was an attacker identified?

Answer 12: To date, there have been no adverse impacts to safety, important to safety, security, or emergency preparedness functions due to cyber attacks at Nuclear Power Plants.

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Session Day and Time: Wednesday, 3:30 pm–5:00 pm

Session Number and Title: W26 Regulatory Challenges in New Reactor Technical Reviews

Session Chair: Lynn Mrowca, Branch Chief, Division of Safety Systems and Risk Assessment, NRC/NRO

Session Coordinator: Daniel Mills, NRC/NRO, tel: (301) 415-1108, e-mail: Daniel.Mills@nrc.gov

Question 1: Is seismic hazard analysis (OBE/SSE) different for new reactors than previous fleet (ex, Vogtle 1 vs. Vogtle 3)?

Answer 1: Yes, the seismic hazard analysis is performed differently for new reactors. It is probabilistic and not deterministic as it was done previously.

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Session Day and Time: Wednesday, 3:30 pm–5:00 pm

Session Number and Title: W27 Status and Path Forward on the Management of Gas Accumulation in Safety-System Piping

Session Chair: Anthony Ulses, Branch Chief, Division of Safety Systems, NRC/NRR

Session Coordinator: Jennifer Gall, NRC/NRR, tel: (301) 415-3253, e-mail: Jennifer.Gall@nrc.gov

Question 1: In the summer of 2008 at the NEI workshop you (Bill Ruland) stood up and said "we'll see!" What have you seen?

Answer 1: In response to the Generic Letter, the licensees' have identified voids in safety related systems and have taken steps to remove or evaluate the impact of such voids. Some of these voids resulted in operation beyond licensing/design basis. Licensees' have identified (and corrected) problems with their filling and venting procedures and practices. Licensees' are performing additional testing/surveillances which are necessary to ensure continued operability of safety systems vulnerable to gas intrusion.

Question 2: Is there or will there be a relationship between NEI 09-10 and the technical specification (TS) section 5 program?

Answer 2: NEI 09-10, Rev 1 provides some excellent insights into how to prevent and manage gas accumulation. The details of a TS section 5 program have not been worked out, but when and if a TS section five program is created it will most likely draw on material from NEI 09-10 Rev 1.

Question 3: What do you mean by "better cooperation needed" to resolve TS issues?

Answer 3: As Bill Ruland said at the end of the session, both NRC and the Technical Specification Task Force (TSTF) must reevaluate their positions and be willing to work with each other in order to produce a solution that protects the public health and safety and both NRC and TSTF accept.

Question 4: How have new reactors incorporated NEI 09-10?

Answer 4: For ESBWR, all Gravity Driven Cooling System (GDCS) and Isolation Condensers system piping will have high point vents. The GDCS will be sloped. For ABWR, there are jockey pumps running all the time to keep the ECCS system pressurized and will have high points. Technical specifications require periodic venting of the high points.

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Session Day and Time: Thursday, 8:30 am–10:00 am

Session Number and Title: TH29 Analysis of Cancer Risk in Populations Living Near Nuclear Facilities

Session Chair: Kathy Halvey Gibson, Deputy Director, Division of Systems Analysis, NRC/RES

Session Coordinator: Vered Anzenberg, NRC/RES, tel: (301) 251-7546, e-mail: Vered.Anzenberg@nrc.gov

Question 1: Have similar studies been done/planned for plant occupational workers who live near power plants vis-a-vis mortality/morbidity? What have the results shown? (Panel)

Answer 1: We are unaware of any studies that explicitly consider the influence of worker risk to the overall population risk around nuclear facilities. As such, the NRC has requested that the Phase 1 committee consider what impact—if any— nuclear workers might have on population risks.

Question 2: Will an option for Phase 2 considerations for NOT going forward be based on low statistical significance and previous NAS (BEIR VII) recommendations to NOT do such studies (because of the inability to distinguish effects of background radiation at 310 mrem to NPP emissions of <5 mrem)?

Answer 2: It is too soon to forecast what the NRC may or may not do in regards to Phase 2. Phase 1 has just begun and the NRC will wait for the results of the study and the subsequent public comments before making a decision on Phase 2. The NRC has requested that NAS determine whether the study request's goals can feasibly be met in a technically defensible way– and if so, develop recommendations for phase 2 using scientifically sound processes for evaluating whether nuclear facilities pose a cancer risk

Question 3: Given the complexity of this analysis and the problems with discerning cause and effect, how should this analysis be described to the public in a manner, which will enhance public confidence?

Answer 3: The NRC believes public confidence will be enhanced by ensuring an open and transparent study process as employed by the National Academy of Sciences.

Dr. Edward Wilds (Connecticut Department of Environmental Protection): The findings should be presented to the public by members of the National Academies Committee that preformed the study at public meetings. This eliminates the appearance that the findings are being filtered by either the NRC or industry. It also give the public a chance to have their concerns answered directly by the experts who conducted the study. I would recommend that during this public presentation piece of the findings that neither the NRC nor industry have any role in the meeting, other than a single NRC individual introducing the committee members present. If anyone other than the committee members participates in any way, it could be viewed as being controlled behind the scenes. NRC and industry should NOT try to answer or clarify any aspect of the study. This needs to be left to the committee members. I also suggest that the NAS provide the moderator or an independent 3rd party moderate the session. The public will generally not be interested in the "numbers" but rather in what this means to "me" and does the committee care about "my safety and health." The public will need to know that the committee members care about their well being first and foremost.

Question 4: How do/will the studies address the increases ability to detect and treat cancers over the past 30-40 years?

Answer 4: The NRC has asked the Phase 1 study committee to consider advances in cancer diagnosis reporting and data collection and how that may be used in a Phase 2 study.

Question 5: How much is NRC paying for this study? Suggest a risk communication document to be issued instead.

Answer 5: The NRC has provided the National Academies a $1,036,653 grant to perform Phase 1 of the study. See answer 2 for the second part of the comment.

Question 6: If the committee finds that phase 2 could be feasible, could NRC staff recommend not to going forward to the Commission based on cost/benefit of doing phase 2?

Answer 6: See answer to Question 2.

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Session Day and Time: Thursday, 8:30 am–10:00 am

Session Number and Title: TH30 Containment Degradation Research and Implications

Session Chair: Mirela Gavrilas, Branch Chief, Division of Engineering, NRC/RES

Session Coordinator: Rasool Anooshehpoor, NRC/RES, tel: (301) 251-7620, e-mail: Rasool.Anooshehpoor@nrc.gov

Question 1: What conditions resulted in the licensee-specific additional inspection activities during the license renewal term?

Answer 1: Whenever an applicant has plant-specific operating experience beyond that covered in the Generic Aging Lessons Learned (GALL) Report, the applicant may need to complete additional inspections to ensure the adequacy of structures during the period of extended operation. Recent examples of this in the structural arena have involved spent fuel pool leakage, reactor cavity leakage, groundwater in-leakage, and containment degradation. Issues like these, beyond the guidance in the GALL Report, are reviewed on a case-by-case basis, and, if necessary, the associated Aging Management Program (AMP) is augmented with the appropriate inspections. Examples of additional activities include concrete core samples to demonstrate that through-wall leakage has not degraded the concrete, and UT samples to demonstrate that leakage has not reduced steel containments or containment liners beyond the minimum wall thickness.

Question 2: With the results of SERF and LERF should we expect an effort in the area of new regulation inspection for SERF and LERF (i.e. aging LERF↓SERF↑)?

Answer 2: The work discussed in the presentation was intended to explore metrics that can be used to evaluate containment degradation in a risk-informed manner. The results of the preliminary scoping study will be used in conjunction with other data to inform inspection practices, as appropriate.

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Session Day and Time: Thursday, 8:30 am–10:00 am

Session Number and Title: TH31 Groundwater Protection

Session Chair: Martin Virgilio, Deputy Executive Director for Reactor and Preparedness Programs, NRC/OEDO

Session Coordinators:
Barry Miller: NRC/NRR, tel: (301) 415-4117, e-mail: Barry.Miller@nrc.gov
Stacie Sakai: NRC/NRR, tel: (301) 415-1884, e-mail: Stacie.Sakai@nrc.gov

Question 1: Does NRC believe that the 3 NEI initiatives will result in prompt remediation of groundwater contamination when prudent? Without new regulations, how can NRC ensure prompt remediation to avoid legacy sites?

Answer 1: The NRC Senior Management Review Group found that the three industry initiatives can, if properly implemented, enhance the prevention, response and remediation of potential threats to groundwater. In addition, the Commission directed the staff to make further improvements to the decommissioning planning process by addressing immediate remediation of residual radioactivity during the operational phase with the objective of avoiding complex decommissioning challenges that can lead to legacy sites. The staff is currently performing a feasibility evaluation and will formulate a recommendation in FY 2011.

Question 2: Why is tritium described in picocuries per liter when most people have no idea what pico stands for? Wouldn't it make more sense to move to per mL and use a more well know prefix for curies?

Answer 2: The unit of picocuries per liter is used because when the concentration of tritium is measured, it is measured in units of picocuries per liter. A picocurie is one trillionth (10^{-12}) of a curie and a curie is a measure of the amount of radioactivity in the material. NRC requirements for reporting information as well as EPA safe drinking water standards use the same units to ensure consistency across the Government. Additionally, the NRC is evaluating how to better communicate information related to tritium to stakeholders so that the information is easier to understand.

Question 3: Will the NRC work with EPA in the potential revision to 40 CFR 190 which will likely include a tritium limit in groundwater applicable to the nuclear fuel cycle (including reactors and 10 CFR 70 fuel cycle facilities and dry storage)?

Answer 3: The NRC is aware of the potential revision to 40 CFR 190 and if the EPA proceeds to revise the regulations will interact with our counterparts in EPA as appropriate during the rulemaking process.

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Session Day and Time: Thursday, 8:30 am–10:00 am

Session Number and Title: TH32 International Cooperation on New Reactors

Session Chair: Gary Holahan, Deputy Office Director, NRC/NRO

Session Coordinator: Donna Williams, NRC/NRO, tel: (301) 415-1322, e-mail: Donna.Williams@nrc.gov

Question 1: What is the influence of MDEP on the current domestic licensing of the EPR and the AP1000 in the countries where these reactors are to be built?

Answer 1: The regulatory authorities that are actively reviewing, preparing to review, or constructing the AP1000 and EPR participate in the MDEP design specific working groups. The working groups share information on a timely basis and cooperate on specific reactor design evaluations and construction oversight. Discussions among the members and sharing of information help to strengthen the individual conclusions reached. In addition to organizing working groups, MDEP has provided each regulator with peer contacts who share information, discuss issues informally, and disseminate information rapidly. For example, the design specific working group members have benefitted significantly from the sharing of questions among the regulators, resulting in more informed, and harmonised, regulatory decisions.

Question 2: The primary founding principle of MDEP was to enable any regulatory safety authority to accept/adopt the safety design of a new reactor as reviewed by MDEP. How close is MDEP to achieving this objective?

Answer 2: The current goals of MDEP are to enhance multilateral cooperation within existing regulatory frameworks, and increase multinational convergence of codes, standards, guides, and safety goals. A key concept throughout the programme is that MDEP will better inform the decisions of regulatory authorities through multinational cooperation, while retaining the sovereign authority of each regulator to make licensing and regulatory decisions. Therefore, there is no current initiative within MDEP to achieve the objective of enabling safety authorities to adopt a review that is approved by MDEP.

Question 3: Is MDEP considering the creation of “observer” membership for countries embarking on nuclear programs? If so, can those observers participate in issue-specific working groups?

Answer 3: Participation in MDEP is intended for mature, experienced national safety authorities of interested countries that already have commitments for new build or firm plans to have commitments in the near future for new reactor designs. MDEP does not have an observer status, but has recently added two new levels of MDEP membership for countries embarking on nuclear programs. The MDEP associate member will be a national regulatory authority without previous licensing experience that has been invited by the MDEP Policy Group to participate in selected MDEP design-specific activities based on evidence that the organization is actively involved in new reactor design review activities relevant to MDEP. Such a regulatory authority would be from a country that has taken a firm commitment in the near term to proceed with safety design review activities, has proprietary agreements with the vendor, and is willing and ready to contribute to specific MDEP activities. It is expected that the associate member would be in a position to exchange information with MDEP members to enhance information sharing and experience in relevant design safety reviews. Associate members would not participate in issue specific working groups.

The second new category of membership is MDEP candidate. This category is intended for countries that have an experienced nuclear regulatory organization, are already regulating nuclear power plants, and have mid- to long-term plans to pursue new reactor licensing and construction. Such regulators could clearly benefit from interacting now with MDEP and, in the near future, could be clear candidates to become MDEP members or associate members. These regulators would participate in the issue-specific working groups of MDEP.

Question 4: How are the MDEP common positions shared with the industry?

Answer 4: The common positions are shared with the industry, and the public, via the MDEP page on the NEA website. After approval of the common position by the working group and the Steering Technical Committee, a common position is posted on the website. The public, including industry, may provide comments to the MDEP through NEA. In addition, representatives of industry and standards development organizations often participate in the working group meetings as observers and maintain an awareness of the common positions as they are under development.

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Session Day and Time: Thursday, 8:30 am–10:00 am

Session Number and Title: TH33 NRC and Federal Incident Response during Real World Events and Exercises

Session Chair: Scott Morris, Deputy Director for Incident Response, Division of Preparedness and Response, NRC/NSIR

Session Coordinator: Sara Mroz, NRC/NSIR, tel: (301) 415-1692, e-mail: Sara.Mroz@nrc.gov

Question 1: Do the agencies consider scenarios such as cyber attacks and electromagnetic pulse events in exercise development?

Answer 1: Exercises are designed to encompass a variety of scenarios and potential events, including cyber attacks and electromagnetic pulse events.

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Session Day and Time: Thursday, 8:30 am–10:00 am

Session Number and Title: TH34 Reactor Oversight Process Reliability

Session Chair: Cynthia Pederson, Deputy Regional Administrator, NRC/RIII

Session Coordinator: William Cartwright, NRC/NRR, tel: (301) 415-8345, e-mail: William.Cartwright@nrc.gov

Question 1: The treatment of cross cutting aspects seems to be inconsistent across the regions. Has this been looked at, and if so what is in place to minimize inconsistency? Program specific issue.

Answer 1: We have not observed any significant inconsistencies in the assignment of cross cutting aspects across the regions. Nevertheless, as noted in the Calendar Year 2009 Reactor Oversight Process (ROP) Self-Assessment, the staff had committed to explore ways to use cross-regional experience to further improve the implementation of guidance on substantive cross-cutting issues (SCCIs). The staff leveraged ongoing efforts initiated by the regions to improve the reliability of ROP implementation, including the SCCI process. The regions are continuing to implement these initiatives, with NRR support.

Additionally, NRC Inspection Manual Chapters (IMCs) 0305, 0612, and 0310 establish guidance for consideration of a licensee's efforts and progress in addressing cross-cutting themes and identifying SCCIs and assessing findings for cross-cutting aspects. The IMCs and inspection procedures are updated periodically to enhance consistency and clarity. Additionally, the ROP undergoes continuous self-assessments and alternating biennial internal and external surveys, which can be used to inform changes to improve guidance application.

As the total number of substantive cross cutting issues across industry is small, it would be statically challenging to make valid objective comparisons between the regions. To make such comparisons, we believe the performance inputs between the licensees and plants in the various regions would also have to be considered. Even if it could be done, the outcome of such an exercise would likely be as contentious as the assignment of substantive cross cutting issues themselves without the benefit.

Question 2: For a number of years the number of violations issued in Region IV has been significantly higher than the other regions. What if anything is this initiative doing to evaluate/address this?

Answer 2: While we have not validated your data, it is an area we could consider as we continue our reliability initiatives. In addition, related to the issue of findings, the quarterly conference calls between Headquarters and the regions, which covers the criteria and threshold of minor and more than minor issues, touches upon the issue of consistent use of our guidance for determining findings. Moreover, the inspector exchanges among the four regions will help reveal any differences in approaches associated with minor versus more than minor determinations and other inspection issues.

Question 3: When the assessment letters are sent, they include an inspection plan. Are the inspection resources listed in the inspection plan based upon the estimates in the IP or what you actually plan to send?

Answer 3: The inspection resources listed reflects the number of inspectors the region plans to send to perform the subject inspections and are based initially on the time estimates specified in the inspection procedure (IP). Sometimes the number of inspectors will be over the estimates in the IP when there is a need for additional inspection focus. For example, if a substantive cross cutting issue (SCCI) has been open for multiple assessment cycles, we may perform additional inspection to assess progress in resolving the SCCI.

Question 4: Why do all security inspections need to be treated as Safeguards or SUNSI? This limits operating experience (OE) opportunities. Can this be changed?

Answer 4: To fully and clearly document an issue, security inspection reports sometimes include security program information that is specific enough to be treated as safeguards information. Our security inspection procedures describe how we assess licensees' physical security programs. Whether in a report or an inspection procedure, the level of detail could potentially be exploited by a would-be adversaries; thereby posing a threat to public health and safety. To prevent that from occurring, shortly after the 911 terrorist attacks, the Commission established a policy to restrict all details of a licensee's physical security program from the public domain, consistent with how safeguards information has always been protected. Therefore, all security inspection reports are designated as either "Safeguards Information" or "Official Use Only – Security-Related Information."

However, this restriction does not preclude OE opportunities. Entities within the nuclear industry share NRC inspection findings among themselves to facilitate OE initiatives. The NRC also has an OE program. Further, for a significant inspection finding with generic implications, the NRC will communicate that information to the nuclear industry, as appropriate.

Question: Can this be changed? Answer: We do revisit this policy periodically. Currently, it remains in effect at the direction of the Commission.

Question 5: Has any thought been given to quantifying the level of consistency and/or variance in ROP implementation?

Answer 5: The challenge here would be to separate process consistency from licensee performance. As licensee performance varies, you would expect different results.

At the present time, we do not plan to attempt to quantify the level of consistency and/or variance in Reactor Oversight Process (ROP) implementation across regions or individual plants. We recognize and expect a certain level of variance in ROP implementation based on unique circumstances. Nevertheless, the ROP is designed to be predictable and repeatable, and we strive to be as consistent as practicable. As we noted in our annual ROP self-assessment, we plan to continue the ROP reliability initiatives and the efforts of the safety culture working group to further improve the ROP, including the substantive cross-cutting issue process and its implementation.

Question 6: Where would one go to find the specific objectives, methods, and outcomes of the various ROP reliability initiatives? Are the initiatives proceduralized? Are the results documented?

Answer 6: The overall goal of the reliability initiatives was to find and reduce any significant process variability between the regions in how they handle oversight activities. Some of that is subjective and would be difficult to quantify. Basically, the methods we are using help determine where the regions have process differences, assess whether those differences have an impact on the results achieved from those processes, and, as appropriate, recommend changes.

We have not captured the initiatives in formal procedures or policies. Periodically, we review the initiatives and revise them to reflect lessons learned during implementation and as other focus areas are found. If the outcome of the initiatives results in recommendations to change to our publically-available program documents (e.g., IMC 0612 or IMC 0305), we process those changes in accordance with established procedures for ROP changes.

Question 7: Heard a lot of process and activity. What did you learn and what changes did you make? Where were there safety gaps? Where were you spending too much effort with too little result?

Answer 7: To date, we have not identified any safety gaps. One of our overall conclusions was that there is a high degree of process reliability between the regions. This is due to a number of factors, including: The inspection documents used are centralized, the ROP process is fairly mature, and there is substantial communication between the regions and Headquarters.

Where we noted inconsistencies, we used the information as input into various ROP self-assessment activities. In addition, assessing the efficiency of the ROP process is considered in the ROP realignment process performed every two years under Inspection Manual Chapter 0307.

We are now in the process of the 2011 ROP realignment. This year, in addition to reviewing the ROP inspection procedures against established review criteria that judge the effectiveness of each procedure, we established several focus areas. The focus areas were identified as part of a recent self-assessment and in response to industry events. At the end of this year’s process we will revise procedures to enhance their effectiveness and will correspondingly realign resources.

We also performed a gap analysis in 2010 with the goal of revealing potential ROP areas that may warrant changes. In our draft gap analysis, we identified one area of potential near-term change in the Public Radiation Safety Cornerstone regarding the ROP's ability to address licensee initiatives in monitoring and controlling releases to groundwater. We will seek Commission policy direction before changing the ROP in this cornerstone.

Question 8: Comment: Good Initiative. Minor threshold should continue to be a focus. Question: How does the assessment explain cases where the plant with the least number of findings in Regions IV would have the highest number if it were in Region I?

Answer 8: We are not aware of any data that supports the case presented in the question. Nevertheless, we plan to continue to look at the application of the minor threshold and overall ROP reliability across the regions. See also, our answer to Question 2.

Question 9: It seems that inspection is encouraged to find problems more and more in general. Is it true in the US that a good inspector is a person who finds more problems?

Answer 9: We place emphasis on our inspectors' abilities to assess licensee performance and identify problems if they exist, not on the number of findings.

Question 10: What ROP areas do you believe the most significant challenge to the operating stations? Why?

Answer 10: Passive systems and equipment that typically do not get much attention and making changes to plants using currently accepted standards to preserve acceptable safety margins.

Question 11: Resource Sharing. In optimizing resource sharing for ROP implementation activities (e.g. inspections and SRA support), are there considerations for increased use of HQ/NRR staff who continue to maintain their inspector credentials?

Answer 11: Regional inspectors have a wealth of experience and technical expertise is virtually all areas of plant operations and departments. Nevertheless, the regional offices request the assistance of HQ and NRR staff who are qualified inspectors and/or technical experts when the need arises. This practice is expected to continue and is dependent on the specific needs of the regions for specific inspections and the availability of qualified headquarters staff with the required experience and technical expertise. As an example, headquarters has several individuals who have been qualified through a safety culture assessor qualification program to participate in safety culture assessments and supplemental inspections.

Question 12: How does the ROP work with COL during construction? When does it transition to the ROP?

Answer 12: Details of that transition are currently being developed. Construction activities obviously require different oversight processes than operational plants. In addition, there is a substantial difference in the potential safety impact to the public from a plant under construction and one that is operational.

As noted in SECY-10-0140, "Options for Revising the Construction Reactor Oversight Process [cROP] Assessment Program," the staff anticipates that the transition from construction to operating reactor oversight would occur once the Commission makes a positive determination under 10 CFR 52.103(g) that all inspections, tests, analyses, and acceptance criteria have been met. The exact timing of the transition and the details have not yet been determined. We plan to continue with the current cROP working group and/or form a new working group that focuses on the transition to the ROP after the Commission has issued its Staff Requirements Memorandum in response to SECY-10-0140.

Question 13: Recently there have been a number of industry issues such as the HB Robinson event that have challenged safety significant systems. 1) Have there been any findings from assessment that would facilitate ID of precursors? 2) How is the assessment process being changed as a result of these events?

Answer 13: As part of the ROP process, the HB Robinson event was assessed to determine if the licensee also had substantive cross cutting issues. Substantive cross cutting issues have the potential to impact multiple safety cornerstones, and could be viewed as potential precursors to more significant problems. The staff has not incorporated other precursors into the Reactor Oversight Process (ROP) to date because precursors that are performance-based and risk-informed have not been identified. The ROP is assessed annually for potential changes, and precursors may be incorporated if appropriate.

Lessons learned from significant events are factored in to various ROP self assessment activities. The staff is currently developing lessons learned from the HB Robinson event in parallel with the 2011 ROP realignment efforts to correct any inspection program weaknesses. The staff is also assessing feedback from the HB Robinson event reactive inspection for possible incorporation into the existing reactive inspection procedures.

Question 14: What can RUGs and NRUG do to make our interactions more effective? How do we avoid being redundant to the other forums?

Answer 14: We all have our responsibilities, and should proactively seek out opportunities to improve our efforts to meet those responsibilities. We suggest that the RUGs and NRUG adopt the attitude of "what can we do to help resolve the problem" and be more proactive in seeking out and using the wealth of information that has been developed by the other forums to resolve issues. Raising the same issues in different forums while expecting different results serves neither the regulator nor the industry. Moreover, pushing back without first fully understanding and assessing the issue at hand usually does not resolve issues in a constructive manner and can be counterproductive.

Question 15: Follow-up to the question on event follow-up: How do you address issues that should have been identified by baseline inspections, that are indicative of regional thresholds, etc., vs. OE, etc.? Example: Robinson AIT identified issues that should have been previously identified in baseline - NRC performance vs. licensee performance.

Answer 15: The NRC performs reactive inspections to further asses the significance of events or degraded conditions. One such reactive inspection is directed by NRC Inspection Procedure (IP) 93800, "Augmented Inspection Team [AIT]." This IP directs inspectors to use the Reactor Oversight Process (ROP) feedback form process in accordance with Inspection Manual Chapter 0308 to provide recommended changes to ROP baseline inspection procedures in order to proactively identify similar issues and causes associated with the event. The staff is developing lessons learned from the NRC's AIT for the Robinson event in parallel with the 2011 ROP realignment for program weaknesses or gaps. One focus of the 2011 ROP realignment is operator qualification and training. A ROP feedback form was submitted as a result of the Robinson augmented inspection. The staff is currently evaluating the feedback for possible incorporation into existing ROP inspection procedures.

The NRC is a learning organization. The ROP has a number of information capture and self assessment activities that are performed to capture performance gaps, and recommend changes to the program when appropriate. These include inspection feed back forms, Operational Event Smart Samples, monthly meetings with stakeholders, an annual self assessment, external stakeholder surveys, ROP inspection realignment activities, and an annual report to the Commission.

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Session Day and Time: Thursday, 8:30 am–10:00 am

Session Number and Title: TH35 Status of Research Activities in Preparation for Licensing of Advanced Reactors

Session Chair: Michael Scott, Branch Chief, Division of Systems Analysis, NRC/RES

Session Coordinator: Kimberly Tene, NRC/RES tel: (301) 251-7533, e-mail: Kimberly.Tene@nrc.gov

Question 1: What ASME Codes and Standards (new or modification) are needed to support the iPWRs?

Answer 1: At this point, NRC does not have the design details of iPWRS. Based on the preliminary design information available, it appears that in general ASME Codes will be applicable. However, some ASME Codes modification may be needed because of the unique design features of iPWRs.

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Session Day and Time: Thursday, 10:30 am–12:00 pm

Session Number and Title: TH36 Containment Accident Pressure and Adequate Net Positive Suction Head

Session Chair: Robert Dennig, Branch Chief, Division of Safety Systems, NRC/NRR

Session Coordinator: Ahsan Sallman, NRC/NRR, tel: (301) 415-2380, e-mail: Ahsan.Sallman@nrc.gov

Question 1:
Given containment cracks at Fitzpatrick and Hatch 1 & 2:
Given containment OD to ID holes at Beaver Valley etc.:
Given containment ID to OD holes at Turkey Point etc.:
Given containment coating failures at Oconee etc.:
Given containment ASME visual inspection failures at Turkey Point etc.:
Given that the NRC has acknowledged it has no comprehensive database on containment failures nationwide:
Please explain why the staff assumes there is ZERO probability of containment leakage when applying the NPSH credit to uprating BWR's.

Answer 1: Containment pressure for crediting to NPSH available is conservatively determined as a minimum value. These calculations assume a containment leakage rate of at least the allowable accident leakage rate specified in the corresponding reactor's technical specifications. Periodic primary containment integrated leakage rate testing (ILRT) verifies leakage potential to be less than the allowable leakage specified in the technical specifications. Containment penetrations, which have historically been the pathways of most containment leakage detected, are tested more frequently and a running summation of their contribution maintained for each containment. Also, for the containment barrier other than testable penetrations, periodic visual inspections in accordance with ASME Boiler and Pressure Vessel Code provisions are made between ILRT performances. These inspections have identified discrete points of degradation with leakage potential well before the leakage could make a significant contribution to the overall leakage measured during an ILRT. Several instances of containment component degradation have been reported over the past 25 years, and a number of NRC generic communications have been issued that addressed these occurrences. Most of the recent events have involved localized corrosion of the carbon steel liner of concrete containments, some with (liner) through-wall penetration, although with very small measured or calculated leakage. None of the events involved a loss of containment design function, including leak tightness assumed in the dose analyses. The NRC staff continues to monitor the industry response to these events, especially with regard to inspection methods and frequency.

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Session Day and Time: Thursday, 10:30 am–12:00 pm

Session Number and Title: TH39 New Reactors Licensed under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants": Changes during Construction (CdC) and Post-Combined License (COL) Licensing Basis Maintenance

Session Chair: David Matthews, Director, Division of New Reactor Licensing, NRC/NRO

Session Coordinator: Amy Snyder, NRC/NRO, tel: (301) 415-6822, e-mail: Amy.Snyder@nrc.gov

Question 1: **"**Some CdCs are inevitable, while others may have future benefits including safety. Why would NRC or a licensee consider any change that puts the Part 52 process at risk?"

Answer 1: To anticipate, address and provide recommendations for effective processing of licensee plant changes and modifications during the construction period under a Title 10, Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," of the *Code of Federal Regulations* (10 CFR Part 52) combined license (COL), the U.S. Nuclear Regulatory Commission (NRC) is determining the activities that can be performed by licensees during construction while the NRC is reviewing requested changes to the licensing basis (license amendments); determining if changes should be recommended for the NRC's enforcement policy to provide for enforcement discretion during new plant construction; determining for new plants what revisions to the risk-informed guidance for evaluating changes to the licensing basis should be required and determine the applicability of the 10 CFR 50.59 guidance; and establishing guidance that should be used for evaluating changes to the severe accident design features (VIII.B.5.c) of each design certification rule (construction & operation).

Question 2: **"**A number of New Reactor Applications use AP1000 (14 Units) and ABWR (2 Units) Designs. AP1000, ABWR, AP600 and CE80+ Designs do not contain Environmentally Assisted Fatigue (EAF) considerations. EAF has been around for over 15 years. All renewed plants and new reactors of ESBWR, USEPR and US-APWR Design have EAF considerations. Why is it acceptable that AP1000 and ABWR Designs do not consider EAF in accordance with RG 1.207 when so may of the new scheduled reactors will utilize these designs?"

Answer 2: The environmental effects on fatigue were addressed in all design certifications and discussed in the NRC staff's final safety evaluation reports (FSERs) since the Advanced Boiling Water Reactor. The NRC staff's evaluation of the environmental effects on fatigue is discussed for each of these standard plant designs in the staff's FSERs in Section 3.12.5.7:

- ABWR (NUREG-1503, "Final Safety Evaluation Report Related to the Certification of the Advanced Boiling-Water Reactor," U.S. Nuclear Regulatory Commission, Washington, DC, July 1994)
- System 80+ (NUREG-1462, NUREG-1462, "Draft Safety Evaluation Report related to the Design Certification of the System 80+ Design," U.S. Nuclear Regulatory Commission, Washington, DC, September 1992.)
- AP600 (NUREG-1512, NUREG-1512, "Final Safety Evaluation Report related to Certification of the AP600 Standard Design," U.S. Nuclear Regulatory Commission, Washington, DC, September 1998.)
- AP1000 (NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," U.S. Nuclear Regulatory Commission, Washington, DC, dated September 2004.)

All these standard plant designs were evaluated by the NRC staff before RG 1.207, "Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components due to the Effects of the Light-Water Reactor Environment for New Reactors" was issued in March 2007 and utilized an approach that was deemed appropriate at that time as the concerns of environmental effects on fatigue were evolving." RG 1.207 states that the guidance only applies to new plants and that no backfitting is intended or approved in connection with its issuance.

Question 3: "Please describe the relationship between Korea Electric Power Company (KEPCO) and Korea Hydro Nuclear Power (KHNP) and Doosan"

Answer 3: The relationship among KEPCO, KHNP and Doosan is as follows: KHNP is a sub-tier company of KEPCO; Doosan is a contractor.

Question 4: "One of the significant factors in the number of CdCs is the quantity of Tier 2* designated descriptions in the DCD. As one involved in the first wave of design certifications, it's my observation that the number of Tier 2* designations has significantly proliferated. Please comment on whether this is the case and, if so, the reason(s)."

Answer 4: The number of Tier 2* designations are determined on a design-specific basis.

Question 5: "The majority of ITAAC items are post-construction: component inspection and test/system pre-op test. Has there been an evaluation of which ITAAC's are likely LAR or PAR versus *10 CFR 50.59* type resolution?"

Answer 5: No. The need for a license amendment is caused by the need for a design change. All Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) will be verified during the construction phase of a COL and any ITAAC change requires a NRC approved license amendment prior to the implementation of that change. The 10 CFR 50.59 process is an evaluation process described in the federal regulations to determine if NRC approval of a change is required prior to the implementation of that change. ITAAC are Tier 1 information in the licensing basis of the plant and as such require a license amendment to change. The proposed CdC Preliminary Acceptability Review process, as an elective part of the License Amendment Request process, would establish that the NRC has no objection to the licensees' installation and testing, at their own risk, of the structures, systems, and components subject to the ITAAC during the NRC review of the license amendment request affecting the ITAAC.

Question 6: "How is the licensee communicating with the NRC on planned changes (LARs) in advance? What are your expectations for an expedited review and what things are being done to provide the information needed for the LAR review? How are you planning to manage these change requests when you are close to finishing the construction?"

Answer 6: Since 2007, the NRC asks applicants annually, through a regulatory information summary, to identify anticipated new licensing activities two or more years in advance to help the NRC with its budget formulation process. Furthermore, during routine interactions with Part 52 applicants, the NRC staff encourages applicants to communicate as soon as possible, any anticipated licensing activities with the NRC. When licensees get close to finishing construction, the NRC will prioritize the necessary licensing actions.

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Session Day and Time: Thursday, 10:30 am–12:00 pm

Session Number and Title: TH41 Security—Kinetic Change

Session Chair: Clay Johnson, Branch Chief, Division of Security Operations, NRC/NSIR

Session Coordinator: Alan Shropshire, NRC/NSIR, tel: (301) 415-0098, e-mail: Alan.Shropshire@nrc.gov

Question 1: "Through the Public media, the Public is worried about EMP [electromagnetic pulse]. (a) What could be said to calm the fears? (b) From the technical point of view, how does the NRC address the issue?"

Answer 1a: The NRC recently updated an assessment of the electromagnetic (EM) vulnerabilities of nuclear power plant digital safety systems. This assessment indicates that the digital safety systems would not be adversely affected to the level where they could not perform their safety function.

Answer 1b: Quality design processes to protect nuclear plant digital safety systems from electromagnetic interference and general construction techniques (such as required for potential seismic events) are expected to provide an adequate level of protection to the digital safety systems from fields that would be generated by an EM event.

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