U.S. Nuclear Regulatory Commission
22nd Annual Regulatory Information Conference
March 9-11, 2010

TECHNICAL SESSION QUESTIONS AND ANSWERS

Please click on the following session numbers to be taken directly to the technical session's questions and answers:

T1	W8	W15	TH22	TH28	TH34
T2	W9	W16	TH23	TH29	TH35
T3	W10	W17	TH24	TH30	TH36
T4	W11	W18	TH25	TH31	TH37
T5	W12	W19	TH26	TH32	TH38
T6	W13	W20	TH27	TH33	TH39
T7	W14	W21			TH40

Technical sessions by date and time of appearance in conference program:

Session Day and Time: Tuesday, 3:00 p.m. – 5:00 p.m.

Session Number and Title: T1 Containment and Reactor Coolant System Pressure Boundary Materials Degradation

Session Co-Chairs: Patrick Hiland, Director, Division of Engineering, NRC/NRR, and Michael Case, Director, Division of Engineering, NRC/RES

Session Coordinators:
Kerby Scales, NRC/NRR, tel: (301) 415-1369, e-mail: Kerby.Scales@nrc.gov
Matthew Kerr, NRC/RES, tel: (301) 251-7968, e-mail: Matthew.Kerr@nrc.gov

All questions raised were answered during the session.

Session Day and Time: Tuesday, 3:00 p.m. – 5:00 p.m.

Session Number and Title: T2 Fatigue Management at Nuclear Power Facilities – Lessons Learned During the Implementation of 10 CFR Part 26, Subpart I

Session Chair: Kamishan Martin, Human Factors Engineer, Division of Inspection and Regional Support, NRC/NRR

Session Coordinators:
Kamishan Martin, NRC/NRR, tel: (301) 415-3469, e-mail: Kamishan.Martin@nrc.gov
Undine Shoop, NRC/NRR, tel: (301) 415-2063, e-mail: Undine.Shoop@nrc.gov

All questions raised were answered during the session.

Session Day and Time: Tuesday, 3:00 p.m. – 5:00 p.m.

Session Number and Title: T3 International Coordination with Countries Pursuing New Nuclear Power

Session Chair: Margaret Doane, Office Director, NRC/OIP

Session Coordinator: Danielle Emche, NRC/OIP, tel: (301) 415-2644,
e-mail: Danielle.Emche@nrc.gov

All questions raised were answered during the session.

Session Day and Time: Tuesday, 3:00 p.m. – 5:00 p.m.

Session Number and Title: T4 New Reactor Licensing Challenges – Past, Present and a View to the Future

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

All questions raised were answered during the session.

Session Day and Time: Tuesday, 3:00 p.m. – 5:00 p.m.

Session Number and Title: T5 NRC Alternative Dispute Resolution

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

Session Coordinator: Shahram Ghasemian, NRC/OE, (301) 415-3591,
e-mail: Shahram.Ghasemian@nrc.gov

Question 1: What percentage of settlements is through "licensee equivalent ADR"?

Answer 1: For calendar year 2007 cases, 2 of 10 settlements were the result of licensee sponsored ADR. For calendar year 2008 cases, 4 of 19 settlements were the result of licensee sponsored ADR and for calendar year 2009 cases, 2 of 7 settlement agreements were the result of licensee sponsored ADR settlement agreements (1 case pending).

Question 2: What is the average settlement amount? Is there a trend across fiscal years?

Answer 2: The NRC does not generally have access to such information because the monetary information, to the extent there is a monetary provision in the settlement agreement, is typically redacted given the confidential nature of such information.

Session Day and Time: Tuesday, 3:00 p.m. – 5:00 p.m.

Session Number and Title: T6 Practical Aspects of Decommissioning Funding Assurance

Session Chair: Timothy McGinty, Director, Division of Policy and Rulemaking, NRC/NRR

Session Coordinators:
Thomas Fredrichs, NRC/NRR, tel: (301) 415-7951, e-mail: Thomas.Fredrichs@nrc.gov
Michael Dusaniwskyj, NRC/NRR, tel: (301) 415-1260,
e-mail: Michael.Dusaniwskyj@nrc.gov

All questions raised were answered during the session.

Session Day and Time: Tuesday, March 9, 3:00 p.m. - 5:00 p.m.

Session Number and Title: T7 Security of Digital Systems

Session Chair: William Kemper, NRR and Scott Morris, NSIR

Session Coordinator:
Samir Darbali, NRC/NRR, tel: (301) 415-1360, E-mail: Samir.Darbali@nrc.gov
Alan Shropshire, NRC/NSIR, tel: (301) 415-0098, e-mail: Alan.Shropshire@nrc.gov

Question 1: There is a lot of concern about EMP. Are we sure that we are protected against such an attack?

Answer 1: Previous research conducted for the NRC in the early-1980s time frame concluded that there was a reasonable assurance of protection afforded to the general types of electrical and electronic equipment in use at that time, due in part to the density of materials and selection of the types of concrete and steel construction typically found in nuclear power plant control/auxiliary buildings and reactor buildings. At present, the NRC is nearing the completion of an update to the previous research to determine if such construction materials would still provide sufficient attenuation of potential EMI fields generated by man-made acts and sufficiently protect safety systems given the known and projected use of significantly more advanced and diverse components (i.e., highly sensitive lower-voltage digital technology.) This research is also intended to identify possible methods (including key design characteristics) as well as areas of additional research to further enhance the protection of safety systems from all EMP effects. Additionally, the NRC is collaborating with other government agencies working to protect or mitigate the impact of EMP on the Critical National Infrastructures.

Question 2: With the upcoming Revision 3 to Regulatory Guide 1.152, does that mean that only Regulatory Guide 5.71 will provide guidance for cyber security programs?

Answer 2: Yes. Regulatory Guide 1.152 will no longer be used cover any aspects of cyber security.

Question 3: Is the (Regulatory Guide 1.152) vulnerability assessment requirement covered by risk assessment under NIST 800 standard referenced by Regulatory Guide 5.71?

Answer 3: At this time, the NRC staff has not evaluated the NIST standards for their applicability to Part 50 licensing.

Question 4: Is the tampering issue discussed in Regulatory Guide 1.152 section Regulatory Position 2.1 also covered by Regulatory Guide 5.71 and referenced NIST 800 standards?

Answer 4: Tampering is not specifically mentioned in Regulatory Position 2.1 of the current revision of Regulatory Guide 1.152. The next revision to the Regulatory Guide is being issued to clarify that Part 50 does not address intentional malicious acts.

Question 5: Since the staff is separating the engineering review under Part 50 from the cyber review under Part 73, does that mean that the staff does not expect cyber controls to be implemented through engineered solutions?

Answer 5: Licensees and vendors have always been able to implement cyber security protection features through design of systems. The adequacy of those controls from a cyber security perspective will be evaluated as part of the 10 CFR 73.54 inspection program. From a Part 50 licensing perspective, the staff is focused on ensuring that any such features do not present a potential for degrading the reliable operation of a safety system or accomplishment of a safety function.

Question 6: Protection against an event such as the Browns Ferry datastorm as an accident would come under NRR, but protection against the same thing as a deliberate intrusion would come under NSIR. Please explain how it makes sense to address a single phenomena under one office or the other on the basis of “intent”.

Answer 6: The impact of equipment failures and inadvertent actions by operators may be predictable and static. For example, once the failure modes of a particular piece of equipment are identified and designed against, one can reasonably expect that challenge to plant safety to be addressed. These types of challenges can be addressed in a licensing action. By contrast, cyber security threats imply an intelligent, less predictable and evolving adversary. The cyber controls of today may be obsolete after a period of time. Constant vigilance is required to address these challenges. Thus, a cyber security program is required under Part 73.

Question 7: How are safety plant controls separated in this digital age?

Answer 7: Clause 5.6.3 of IEEE 603-1991, which is endorsed by 10 CFR 50.55a(h), and General Design Criterion 24 of Appendix A to 10 CFR 50 both address independence of safety systems from other plant equipment. These requirements are applicable to digital systems.

Question 8: Were the security controls selected by NRC staff from NIST SP 800-53 based on High values across the board for Confidentiality, Integrity, and Availability?

Answer 8: Yes. The High baseline security controls contained in NIST SP 800-53 were evaluated and then tailored to address protections for Confidentiality, Integrity, and Availability of critical plant systems performing safety, security, and emergency preparedness (SSEP) functions. In close coordination with industry representatives, NRC staff followed the NIST SP 800-53 tailoring process to ensure security controls contained in Regulatory Guide 5.71 are sufficient to protect critical digital assets performing SSEP functions from cyber attacks, up to and including the design basis threat as described in 10 CFR 73.1.

Question 9: When selecting the 148 security controls contained in Regulatory Guide 5.71, what level of effort went into ensuring that the recommended security controls are truly applicable to digital control systems versus federal information systems? For example, what considerations were made to ensure implementation of the security controls will not disrupt plant operations?

Answer 9: The security controls defined in Regulatory Guide 5.71 are the result of an extensive tailoring process carried out by NRC staff, with extensive input from experts in the field of cyber security, the Pacific Northwest National Laboratory, representatives from the Nuclear Energy Institute, and licensing experts and cyber security practitioners from within the nuclear power industry. The tailoring process conformed with NIST SP 800-53 standards and was used to ensure each security control in Regulatory Guide 5.71 addresses the unique requirements of systems performing safety, safety-related, or emergency preparedness (SSEP) functions at the plant. Furthermore, guidance contained in Regulatory Guide 5.71 states that a security control should not be applied if the control could adversely impact an SSEP function or performance at the plant. When a control cannot be fully implemented, provisions in the guidance state that licensees are to either implement alternative controls or countermeasures that provide an equivalent level of protection or provide documented justification explaining why the need for a specific security control does not exist.

Question 10: What considerations are in place in terms of performing audits and inspections for plant systems that fall under FERC Order No. 706-B and are deemed outside the scope of systems under 10 CFR 73.54?

Answer 10: On January 11, 2010 the NRC issued a press release regarding the Memorandum of Understanding (MOU) entered into by the NRC and the North American Electric Reliability Corporation (NERC). The press release states, "The MOU acknowledges the NRC's regulatory responsibility for inspecting digital systems that can affect safety, security and emergency preparedness of a nuclear power plant as well as NERC's responsibility for regulating digital systems related to continuity of electric power generation. As part of the MOU, the NRC and NERC agree to share information discovered during respective inspections that they believe may be relevant to any digital system governed by the other organization."

NRC Press Release is available at http://www.nrc.gov/reading-rm/doc-collections/news/2010/10-005.html.

NERC Press Release is available at http://www.nerc.com/news_pr.php?npr=489.

Question 11: Is the design basis threat (DBT) for cyber security expected to change over time?

Answer 11: The DBT is systematically evaluated and validated yearly by the NRC's Intelligence Liaison and Threat Assessment Branch (ILTAB), Division of Security Operations (DSO), Office of Nuclear Security and Incident Response (NSIR). ILTAB also coordinates with other NRC offices, in addition to law enforcement and intelligence community organizations and agencies, to provide strategic and tactical intelligence warning and analysis of all threats to the U.S. commercial nuclear sector. The Commission meets annually (or as needed) to evaluate the need to modify the DBT.

Question 12: Is the NRC going to accept or reject the cyber security plans submitted by licensees on November 23, 2009?

Answer 12: Licensees' cyber security plans submitted to the NRC on November 23, 2009, utilized the cyber security plan template provided in Appendix A of NEI 08-09, "Cyber Security Plan for Nuclear Power Reactors", Revision 3. The NEI 08-09, Revision 3 is not endorsed by the NRC as an acceptable approach for meeting the requirements of 10 CFR 73.54(e).

The Nuclear Energy Institute (NEI) and Executive Task Force of the industry's Nuclear Security Working Group (NSWG) were selected to represent the operating power reactor licensees when addressing the concerns the NRC staff raised with NEI 08-09, Revision 3. These concerns were documented by NRC staff in for the form of generic requests for additional information (RAIs) and provided to NEI and NSWG by letter which is publically available in ADAMS.

As a result, a number of changes to NEI 08-09, Revision 3, were proposed by NEI and NSWG that are now under review by NRC staff. Acceptance of licensees' cyber security plans is contingent upon the results of these reviews, which are scheduled to conclude in early April 2010.

Question 13: Are there any cyber issues or requirements unique to VPN access for ERDS modernization efforts?

Answer 13: As a federal information system, ERDS is subject to the security requirements set forth under the Federal Information Security Management Act (FISMA) of 2002. The ERDS modernization efforts address maintenance and security limitations associated with the use of dial-up modems for the transmission of plant data to the NRC Operations Center during certain emergency events. Transition from the use of dial-up modems to Internet-based Virtual Private Network (VPN) technology supports compliance with the FISMA security requirements and promotes the use protective measures that make the system more resilient to failure or compromise. Recent information on the NRC position related to ERDS can be found two documents posted in ADAMS. The first document is entitled, Emergency Response Data System and Cyber Security, can be found in ADAMS at ML100130359 and the second, Emergency Response Data System Modem Transition, can be in ADAMS at ML100261085.

Session Day and Time: Wednesday, 10:30 a.m. – 12:30 p.m.

Session Number and Title: W8 Lessons Learned from Sixty Years of Radioactive Waste Management and the Path Forward for the NRC

Session Chair: Michael Weber, Office Director, NRC/NMSS

Session Coordinator: Elizabeth Doolittle, NRC/NMSS, tel: (301) 492-3238, e-mail: Elizabeth.Doolittle@nrc.gov

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

Session Day and Time: Wednesday, 10:30 a.m. – 12:30 p.m.

Session Number and Title: W9 License Renewal Program

Session Chair: Brian Holian, Director, Division of License Renewal, NRC/NRR

Session Coordinator: Session Coordinator: David Drucker, NRC/NRR, tel: (301) 415-6223, e-mail: David.Drucker@nrc.gov

All questions raised were answered during the session.

Session Day and Time: Wednesday, 10:30 a.m. – 12:30 p.m.

Session Number and Title: W10 Licensing Best Practices and Initiatives

Session Chair: Robert Nelson, Deputy Director, Division of Operating Reactor Licensing, NRC/NRR

Session Coordinator: Edward Miller, NRC/NRR, tel: (301) 415-2481, e-mail: Ed.Miller@nrc.gov

All questions raised were answered during the session.

Session Day and Time: Wednesday, 10:30 a.m. – 12:30 p.m.

Session Number and Title: W11 Radiation Protection

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

Session Coordinator: Elijah Dickson, NRC/RES, tel: (301) 251-7519, e-mail: Elijah.Dickson@nrc.gov

Question 1: Are you planning on looking at updating the environmental models and guidance (GASPAR and LADTAP)? These models are used in environmental impact statements for the Combine Operating Licenses and the models are very dated.

Answer 1: These codes have been updated.

Question 2: Has the Agency considered a approach to inform the public – such as communities near nuclear power plants about risks of radiation and health impacts?

Answer 2: Every nuclear power facility has an annual public meeting called the "end-of-cycle" where the agency reviews the safety performance of the plant and he public is offered an opportunity to ask questions of NRC staff. Additionally, the NRC has a fact sheet on radiation health effects at http://www.nrc.gov/reading-rm/doc-collections/fact-sheets/bio-effects-radiation.html

Question 3: Denver, Colorado has the highest natural background dose rate, yet the lowest cancer rate in the country. Is a little radiation actually good for us? How much is?

Answer 3: Along with other national and international regulatory agencies responsible for radiation protection, the NRC assumes that any exposure to radiation poses some health risk, and that risk increases as exposure increases in a linear, no-threshold (LNT) manner. The LNT assumption suggests that any increase in dose, no matter how small, incrementally increases risk. Conversely, lower levels of radiation proportionately decrease the risk, such that very small radiation doses have very little risk.

Question 4: The U.S. Commission spoke of "Excellence" as one of the principles. How can you relate this to the U.S. approach to fall behind the international standards for radiation protection?

Answer 4: Excellence is a key organizational value that guides NRC actions — from decisions on safety, security, and environmental issues, to how the NRC interacts with stakeholders and conducts administrative tasks. In implementing this organizational value, the NRC employs some of the most highly educated, trained, and experienced radiation protection professionals in the industry. The NRC staff participates in the technical committees of the International Commission on Radiological Protection, the United Nations Scientific Committee on the Effects of Atomic Radiation, the National Council on Radiation Protection and Measurements, the U.S. Department of Energy's Low Dose Radiation Research Program, and similar international scientific bodies.

With respect to the NRC's standards for radiation protection, the existing regulatory framework continues to provide adequate protection of the health and safety of workers, the public, and the environment. Currently, the NRC staff is conducting outreach with stakeholders and interested parties to initiate development of a technical basis for possible revision of the radiation protection regulations, as appropriate and where scientifically justified, to achieve greater alignment with international recommendations.

Question 5: If I understand…. The National Cancer Institute has not shared any tracking trend cancer information with your [RES] office. I though NC at Fort Detrick was required to share info with the U.S. Government. Please clarify.

Answer 5: The NRC staff approached NCI management about updating the 1990 study under an inter-agency agreement with the NRC, but because of staffing limitations, NCI was unable to commit staff resources for this activity for the foreseeable future. The cancer mortality and incidence databases needed for the study are available at various federal and state agencies. Finally, we are unaware of an office of the National Cancer Institute at Ft. Detrick.

Question 6: Considering the recent NCRP Report No. 160 with U.S. Medical Dose at 3000 mRem/year. Is NRC considering studying cancer from the [medical] use to validate LNT?

Answer 6: Currently, NRC is not planning to fund a study using the medical radiation doses resulted from procedures performed on patients in the U.S. medical centers to validate LNT.

Question 7: How is NRC coordinating the development of the new Regulatory Guides with other NRR branches that will be evaluating design basis accident doses in TEDE (e.g. NRR/DRA/AADB)? Is it important that the branches who develop guidance communicate with the branches that use it?

Answer 7: Currently, RES is in the process of updating Regulatory Guides 8.7 and 8.34 to align with the definition change of TEDE in 10 CFR Part 20. In addition, NRR coordinated with RES for the review of Regulatory Guide 8.40. Since these guides pertain to recording and reporting external and internal occupational radiation doses for operating nuclear power reactors, RES has not specifically coordinated with NRR/DRA/AADB. However, RES can seek input on Regulatory Guides 8.7 and 8.34 from this group. Yes, RES believes that it is important to coordinate with the appropriate branches across the agency to ensure alignment of these regulatory guides.

Question 8: Is the industry prepared to make concrete proposals to restructure NRC Radiation Protection Regulation and Guidance?

Answer 8: Yes, the industry is prepared to make concrete proposals to restructure NRC Radiation Protection Regulations and Guidance. We are planning to provide this input at the series of public workshops that NRC has announced will be held later in 2010.

Question 9: Will the presidential guidance provisions of the AEA be met? Will there be consistency in the NRC? Will all doses in other Part of 10 CFR be in units of effective dose? Will there be consistency across the federal family? Are the states on board? Do they regulate more users than the NRC? Will NRC alter its software to effective dose?

Answer 9: The NRC staff plans to continue to work with the Environmental Protection Agency, and other Federal Agencies, to achieve a consistent approach to radiation protection. The presidential guidance, commonly known as Federal Guidance, is one of the topics under discussion by the Federal Agencies. The NRC staff is also examining all aspects of the NRC regulations, and has identified a number of areas where requirements are currently specified that are not entirely consistent with the terminology and methodology of 10 CFR Part 20. As agreed upon by the Commission, the current rulemaking efforts focus upon the issues and possible updates to 10 CFR Part 20, and 10 CFR Part 50 Appendix I. Other portions of the regulations will also be examined, but are not presently to be part of the initial rulemaking effort.

For example, the Commission has separately directed the NRC staff to examine a risk informed examination of the requirements for low level waste disposal in 10 CFR part 61. The states, and particularly Agreement States, are important parties to the ongoing examination of the regulations. The NRC staff has interacted with both the Conference of Radiation Control Program Directors, and with the Organization of Agreement States to involve the states in the initial discussions of possible changes and impacts. The NRC staff is aware that many computer programs will need to be updated, if changes are made within the regulations, and is already engaged in examining changes to some of the software packages.

Question 10: Do you foresee to revise 10 CFR Part 20 dose limits to incorporate ICRP's most recent recommendations?

Answer 10: The NRC staff is currently engaging stakeholders to examine the issues of increasing alignment with ICRP's most recent recommendations. The question of dose limits, and in particular occupational dose limits, is one of the key issues under discussions. The NRC staff has received feedback on both sides of the issue, with some stakeholders strongly supporting a change to the dose limits, and other stakeholders not being supportive of a change. The NRC staff will be continuing the dialogue with all stakeholders in preparation for making recommendations to the Commission on a possible path forward.

Question 11: You're slide on the percentage of exposure from various sources was very informative, but do you really mean "members of the public" for doses received from nuclear power plant waste, or do you mean "occupational workers"?

Answer 11: The slide in question presents the repartition of the exposure to ionizing radiation, as averaged for the French population as a whole. Doses from all nuclear installations (not only NPPs) are included in 2 forms: the doses to the workers (currently around 300 000 persons including the medical sector), as recorded from their individual surveillance, and the environmental dose to the population near the sites, as computed through environmental exposure IRSN model using as source term the radioactive releases authorized for each site. Further information may be found at the following link to IRSN web site:
http://www.irsn.fr/FR/expertise/rapports_expertise/Documents/radioprotection/IRSN_Rapport_Exposition_Population_2006.

Recent statistics for occupational exposure for workers, and for patient's exposure through medical diagnostic can also be found on http://www.irsn.fr.

Question 12: Where is the European Low Dose Research published?

Answer 12: As in the US, research results from European scientists are published in different scientific journals according to the disciplines involved. I could quote in particular the following journals which have a good record for publications in this field : Radiation Research, Radiation Protection Dosimetry, Health Physics, Int. Journal of Radiation Oncology Biology Physics. I would add that UNSCEAR (http://www.unscear.org/) regularly analyses at the international level the scientific output in the field of ionizing radiation effects, and draws up substantial periodic reports providing a critical review and synthesis of research results.

The MELODI European platform (http://www.melodi-online.eu/) will eventually also keep track of scientific research production in the field of low dose effects. Finally, the science section of IRSN website also presents the institute's research programmes and key results.

Session Day and Time: Wednesday, 10:30 a.m. – 12:30 p.m.

Session Number and Title: W12 Regulatory and Policy Issues for Small Modular Reactors

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

Session Coordinator: Joseph Williams, NRC/NRO, tel: (301) 415-1470, e-mail: Joseph.Williams@nrc.gov

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

Session Day and Time: Wednesday, 10:30 a.m. – 12:30 p.m.

Session Number and Title: W13 Regulatory Applications of International Operating Experience

Session Chair: Michael Cullingford, Special Assistant, NRC/NRR

Session Coordinator: Karen Zeleznock, NRC/NRR, tel: (301) 415-1416, e-mail: Karen.Zeleznock@nrc.gov

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

Session Day and Time: Wednesday, 10:30 a.m. – 12:30 p.m.

Session Number and Title: W14 Technical Review Strategies for Post-COL Licensing Submittals

Session Chair: Mark Lombard, Deputy Director, Division of Safety System and Risk Assessment, NRC/NRO

Session Coordinator: Tanya Ford, NRC/NRO, tel: (301) 415-1194, e-mail: Tanya.Ford@nrc.gov

All questions raised were answered during the session.

Session Day and Time: Wednesday, 2:00 p.m. – 4:00 p.m.

Session Number and Title: W15 Combined Licenses and Public Accessibility – Current Issues in the Adjudication of Applications for New Reactor Licenses

Session Chair: Alex Karlin, Judge, NRC/ASLBP

Session Coordinator: SherVerne Cloyd, NRC/ASLBP, tel: (301) 415-6504, e-mail: Alex.Karlin@nrc.gov

All questions raised were answered during the session.

Session Day and Time: Wednesday, 2:00 p.m. – 4:00 p.m.

Session Number and Title: W16 Current Topics in Probabilistic Risk Analysis

Session Chair: Kevin Coyne, Branch Chief, Division of Risk Analysis, NRC/RES

Session Coordinator: Michelle Gonzalez, NRC/RES, tel: (301) 251-7591, e-mail: Michelle.Gonzalez@nrc.gov

All questions raised were answered during the session.

Session Day and Time: Wednesday, 2:00 p.m. – 4:00 p.m.

Session Number and Title: W17 Engineering and Regulatory Approaches to New and Innovative Design Submittals

Session Chair: Laura Dudes, Deputy Director, Division of Engineering, NRC/NRO

Session Coordinator: Joel Jenkins, NRC/NRO, tel: (301) 415-2955, e-mail: Joel.Jenkins@nrc.gov

Question 1: Are the current LWR fleet providing public health protection in case of an attack by a large aircraft? If yes, why the new rule for new reactors 50.54(hh) is needed if new reactors can be built to the old LWR fleet standards?

Answer 1: The NRC via its regulation in 10 CFR 50.54 (hh) is helping to assure that in case of an attack by a large aircraft, the current operating reactors provide an adequate protection to public health and safety through the implementation of strategies and measures to mitigate the potential damage consequences..

Specifically, after 9/11/2001, the Commission issued an order to licensees of all operating reactors to develop and adopt mitigative strategies to cope with large fires and explosion from any cause including beyond design basis aircraft impact. These requirements in 10CFR 50.54, conditions of licenses will be able to implement effective mitigation measures through operational actions. Compliance with 50.54 (hh) provides the adequate protection to public health and safety.

The applicant of new reactors would be subject to both the aircraft impact assessment rule and the regulation in 10CFR 50.54(hh) to enhance the plant safety and robustness during design stage and provide the adequate protection to public health and safety.

The goal of the AIA rule in 10 CFR 50.150(a) is to enhance the plant safety and robustness during the design stage to withstand large commercial aircraft impacts, with reduced reliance on operator actions. The Commission regards it as a beyond-design basis rule. The AIA rule is not necessary for adequate protection to public health and safety.

Question 2: In the 1970-80's licensees were subjected to ever evolving rule changes that were costly and caused delays.

(1) It appears that the AIA is just more of the same evolving rules in 2009-2010. Will this continue thru the licensing of new plants?
(2) Will existing plants be back fit with this rule?

Answer 2: The Commission promulgated the AIA rule (10CFR 50.150) to enhance the plant safety and robustness of new reactors during design stage to withstand large commercial aircraft impacts, with reduced reliance on operator actions. The commission regards the AIA rule to be final. The rule is not anticipated to change unless staff determines during the implementation phase that the goals of enhanced safety are not met.

The AIA rule requirements apply only to new reactor designs. The operating plants will not be backfitted with the AIA rule. However, the licensees of all operating plants are required to comply with the regulation in 50.54 (hh) to ensure that nuclear power plants implement effective mitigation measures to cope with large fires and explosion from any cause including beyond design basis aircraft impact through operational actions.

Question 3: ACI 349 prohibits the use of some moment frames unless demonstrated by test & analysis. What is NRC position on test & analysis performed to demonstrate strength & toughness required to resist design loads?

Answer 3: The NRC position on "the subject of performing test & analysis to demonstrate strength & toughness required to resist design loads" is that it will accept conclusions from appropriate testing and properly analyzed test data. For the case of reinforced concrete moment frames, staff notes that ACI 374.1-05, "Acceptance Criteria for Moment Frames Based on Structural Testing and Commentary," provides guidance with respect to design procedure, test modules, test method, and acceptance criteria.

Session Day and Time: Wednesday, 2:00 p.m. – 4:00 p.m.

Session Number and Title: W18 International Activities in Long Term Research Projects

Session Chair: Brian Sheron, Office Director, NRC/RES

Session Coordinator: Donna-Marie Perez, NRC/RES, tel: (301) 251-7518, e-mail: Donna-Marie.Perez@nrc.gov

All questions raised were answered during the session.

Session Day and Time: Wednesday, 2:00 p.m. – 4:00 p.m.

Session Number and Title: W19 Regional Session - Operating Nuclear Power Plant Issues

Session Chair: Mark Satorius, Regional Administrator, NRC/Region III

Session Coordinator: John Jandovitz, NRC/Region III, tel: (630) 829-9763, e-mail: John.Jandovitz@nrc.gov

Question 1: Region III Knowledge Transfer (KT) initiatives were discussed, but it sounded specific to Region III and "other regions are doing some of these things." If these initiatives are best practice, shouldn't they be shared and standardized across the regions?

Answer 1: The agency, including the Regions, all are developing various KT tools tailored to the specific work areas (inspections, licensing). These tools and initiatives are also highlighted within their own intranet sites and highlighted to other offices. It is not uncommon to have staff from other Offices visit the RIII KT website, and vice-versa. There are some KT activities which are “standardized” in that all offices use the agency’s Knowledge Center. The Knowledge center is a web-based system which allows establishment of forums and groups to share information.

Question 2: For dual unit sites, certain program inspections appear to occur more often than necessary. An example is the ISI inspection; we have a program inspection each outage.

Answer 2: The Reactor Oversight Process, specifically the baseline inspections (which includes the ISI inspections), is reviewed every two years to determined it’s effectiveness. Based on this review, the need for each baseline inspection is evaluated as well as the scope and hours suggested for completion of the inspection. Where the inspection is not dependant on the number of units at a site inspection hours contained in the ROP inspection process are not increased over that of a single unit. For other ROP inspections where differences in design, age, and operations between the units can impact the scope of the inspection, such as ISI, inspection hours are increased to account for the multiple units.

Question 3: Has the NRC evaluated the effectiveness and validity of the Significance Determination Process for inspection findings? The SDP sometimes appears too focused on individual components or procedures instead of the overall potential significance of the event or condition.

Answer 3: A self-assessment is performed annually on the Significance Determination Process that solicits input from all stakeholders, including the industry. The results from the self-assessment are reviewed by the NRC to implement improvements as necessary.

Question 4: Do you have plans to address a growing issue with new NRC reviewers struggling to approve License amendments that would not have been an issue in approving in the past (with experienced reviewers)?

Answer 4: The NRC has a qualification process for personnel involved in documents reviews, which included license amendments. Until qualified, the technical reviewer works with a more experienced reviewer who will sign off on the reviewers work. The branch qualification plans ensure that the reviewers are being exposed to the right technical material to help them become proficient in their technical areas, and together with having the peer review inherent in working with the experienced reviewers provides the needed quality.

Question 5: Regarding knowledge transfer, it used to be that following a previously NRC approved "method" for an analysis was acceptable - now we sometimes hear "it doesn't matter if NRC approved it before." Change in NRC position, or the result of new people in role(s)?

Answer 5: There are occasionally cases where inspectors identify issues and concerns regarding licensee design and/or operating practices. If the design, for example, was previously approved and determined acceptable, inspectors cannot impose a change or propose enforcement without further NRC review. It is during this further review, where the NRC may find that the safety issue is valid but yet the licensee is meeting the licensing basis. In these cases, agency must make a determination whether a specific backfit, in accordance with 10CRF 50.109, is appropriate to resolve the safety issue. Licensee's should recognize that although the agency may have approved some facet of licensing basis, that action (e.g., design approval) always remains subject to further NRC review. Concerns raised through that review must then conform to agency process with respect to what if any further action is warranted to resolve safety concerns.

Question 6: Recent information for the staff indicates that Idaho National Labs (INL) is reviewing for Safety System Failures (SSF) and this term is different than a Safety System Functional Failure (SSFF). Please comment on the potential confusion.

Answer 6: INL continues under a long term contract to assess safety system failures (SSFs) for the NRC staff's Industry Trends Program. The definition of an SSF is not fully consistent with the definition of a safety system functional failure (SSFF) under the Reactor Oversight Process (ROP) Performance Indicator (PI) definition. SSFs precede the ROP and is an legacy item that has been maintained intact for long term trending of safety system failures. INL has maintained the SSF database intact for over 20 years, with the same guidance, for purposes of long term trending so that the data is preserved and remains consistent.

When the LER rule and NUREG 1022 were revised in 2000, and with the need to be distinct relative to the implementation of the ROP's Performance Indicator Program (and the SSFF PI) that is tied to 50.73(a)(2)(v) reporting, the term, "SSFF" was born and thus there were two types of system failures the staff was concerned with (SSFs and SSFFs). While the two definitions are very similar in many respects, there are some differences (for example: SSFFs count LOOPs, SSFs do not count LOOPs; Systems that are included under SSFF “scope” include TS-related systems and are defined by the UFSAR, SSF systems are determined from a legacy list of systems specified by AEOD/INL.)

INL does not track or sort on SSFFs, only SSFs. But they do read LERs and sometimes make an assessment of SSFF reporting using their SSF database. The INL Coding Manual also contains a simplified guide to how to interpret NUREG 1022 (a)(2)(v) guidance. This document is used by INL to determine if an LER should have been reported against the (a)(2)(v) criteria, as well as against any other criterion under the LER rule. Currently, INL only tracks SSFs but also provides informal feedback to the agency on observations related to SSFF reportability. Regarding licensee access to the INL “guidance” on SSFF reportability, the agency does not utilize it for determining SSFF reportability or compliance, and it would not be appropriate for the industry to use it either.

As discussed at the R2 RUG meeting, the agency is continuing to examine issues with inconsistent reporting of SSFFs.

Question 7: If a plant is found to have an "unacceptable safety culture," what is the expected NRC response and are there expected actions that the licensee should take?

Answer 7: The NRC does not make determinations that a licensee's safety culture is "unacceptable.” However, routine (baseline) NRC inspections can provide insights into areas important to safety culture, and the NRC may focus inspection effort on reviewing licensees' corrective actions for cross-cutting issues. The NRC also reviews safety culture components as part of its supplemental inspection program and may request licensees to perform safety culture assessments in response to degrading performance that causes the licensee to be in the Degraded Cornerstone or Multiple/Degraded Cornerstone Columns of the ROP Action Matrix. The Reactor Oversight Process (ROP) has many flexibilities available to the staff and includes an action matrix that describes the levels of NRC oversight and inspection based on plant performance.

For example, if a licensee moves into the multiple/repetitive degraded cornerstone column (known as column 4) of the action matrix, the NRC will initiate a supplemental inspection program to investigate the extent of safety culture problems. Further, the NRC may request that the licensee conduct an assessment of its safety culture and develop corrective actions which will be assessed during future NRC inspections. As it deems appropriate, the NRC itself may also conduct in-depth inspections of safety culture at "column 4" sites as part of its supplemental inspection effort. The NRC will continue its increased oversight activities until the licensee can demonstrate that it has improved its processes and performance such that the NRC has renewed confidence in the licensee’s ability to identify and resolve its performance issues.

Session Day and Time: Wednesday, 2:00 p.m. – 4:00 p.m.

Session Number and Title: W20 Siting Safety and Environmental Reviews – Looking Forward

Session Co-Chair: Scott Flanders, Director, and Nilesh Chokshi, Deputy Director, Division of Site and Environmental Reviews, NRC/NRO

Session Coordinator: Carolyn Lauron, NRC/NRO, tel: (301) 415-2736, e-mail: Carolyn.Lauron@nrc.gov

All questions raised were answered during the session.

Session Day and Time: Wednesday, 2:00 p.m. – 4:00 p.m.

Session Number and Title: W21 The Role of the Committee to Review Generic Requirements

Session Chair: James Lyons, Deputy Office Director, NRC/RES

Session Coordinator: Les Cupidon, NRC/RES, tel: (301) 251-7684, e-mail: Les.Cupidon@nrc.gov

All questions raised were answered during the session.

Session Day and Time: Thursday, March 11, 8:30 a.m. – 10:30 a.m.

Session Number and Title: TH22 Buried Piping and High Density Polyethylene Piping

Session Chair: Michele Evans, NRR/DCI

Session Coordinator: Michael Orenak, NRR/DCI, 415-3229, e-mail: Michael.Orenak@nrc.gov

Question 1: Are there any other radioisotopes identified with tritium in the leaking pipes, e.g. transuranic elements? And are the NRC and/or licensee testing groundwater samples for other isotopes?

Answer 1: Other radionuclides have been identified in some leaks at some plants. The types of nuclides present are dependent on the source of the leak. In general terms, the leak sources are either (1) condensed steam, (2) diluted liquid radwaste, or (3) primary systems.

Some leaks involve only condensed steam or condensed water vapor. Typically, the only radionuclide in water vapor is tritium, so these types of leaks are generally limited to tritium only. These types of leaks generally involve extremely small amounts of tritium in very small volumes. Plants typically test for tritium and other gamma-emitting nuclides for these types of leaks.

Some leaks involve diluted liquid radwaste that was in the process of being discharged when the leakage occurred. In these cases, the liquid had already been processed by systems that remove nearly all radioactive materials except tritium, noble gases, and very small amounts of other nuclides. Additionally, these types of leaks have typically been diluted by plant process water. Examples of some of the other nuclides that could be present include Cs-137, Cs-134, Co-58, or Co-60. Because these nuclides are relatively easy to detect, when leaks are discovered, plants routinely conduct tests for these nuclides. This type of leak could also contain other "hard-to-detect" nuclides such as Fe-55, Sr-90, Ni-63, or transuranics, and plants commonly contract with independent laboratories to conduct tests for these nuclides. Hard-to-detect nuclides such as Sr-90, Ni-63, and transuranics are generally not detected in these types of leaks since the water had previously been processed to remove radioactive materials. Plants have developed criteria to determine when it is appropriate to test for transuranics and other hard-to-detect nuclides. For example, plants may elect to test for transuranics and other hard-to-detect nuclides when tritium is detected. To date, NRC's review of these types of leaks indicate transuranics are not typically found in the affected soil or groundwater. It should be noted that tritium, since it is water, travels at the same speed as the groundwater. Other nuclides generally travel much more slowly, and tend to be trapped in the soil in the immediate vicinity of the leak.

Other plants have leaked water from primary systems or auxiliary plant systems that are closely related to primary systems. Examples of these types of leaks would include leakage from spent fuel pools. Salem and Indian Point are two plants that have experienced these types of leaks. These types of leaks are not typically caused by leaking pipes; nonetheless, they have the potential to contain the highest concentrations of tritium and other nuclides. In these cases, plants typically test for tritium, easy-to-detect nuclides, hard-to-detect nuclides, and transuranics. Sr-90, Co-60, and Cs-137 have been detected in these types of leaks. Again, these radionuclides tend to be much less mobile than tritium in groundwater.

Question 2: The recent issues in the press regarding tritium detected in groundwater at Vermont Yankee have raised concerns over the safety of nuclear operations nationwide. What actions is the NRC taking to provide factual information regarding the low risk to public health from tritium present in groundwater at nuclear power plant sites?

Answer 2: The NRC's Office of Public Affairs has updated the agency's fact sheet regarding Tritium, Radiation Protection Limits, and Drinking Water Standards (http://www.nrc.gov/reading-rm/doc-collections/fact-sheets/tritium-radiation-fs.html) and has produced a new fact sheet on Buried Piping at Nuclear Reactors (http://www.nrc.gov/reading-rm/doc-collections/fact-sheets/buried-pipes-fs.html). Both documents, as well as the agency's response to press queries, stress the lack of health impact from the tritium leaks. Additionally, the NRC is hosting a Groundwater Contamination public meeting at headquarters on April 20. The low risk to public health from tritium present in groundwater at nuclear power plant sites will be within the scope of discussion at this meeting.

Question 3: Since NEI initiatives are voluntary, what conditions would need to be met before the NRC would implement enforcement actions for onsite leaks of radioactive material from plant support systems?

Answer 3: The NRC could take enforcement actions if the licensee violated limits established by regulations on radioactive material releases. Additionally, the NRC could take action if the licensee failed to follow its procedures. A licensee not adhering to a self imposed standard is handled through the Reactor Oversight Process (ROP) and, depending on the significance of the event, can potentially result in additional NRC inspections. In addition, the agency can also take enforcement actions, such as issuing an Order, if an apparent safety issue is identified.

Question 4: (to Chairman Jaczko on 3/9/10) Since the tritium leaks from buried piping is of low safety significance, why not use the IAEA's International Nuclear Event Scale (INES), of which NRC participates with IAEA and 60 other countries, to communicate the low safety significance of these events to the public?

Answer 4: It is the policy of the U.S. Nuclear Regulatory Commission to participate in the International Nuclear Event Scale (INES), jointly developed by the International Atomic Energy Agency (IAEA) and the Nuclear Energy Agency of the Organization for Economic Cooperation and Development (NEA/OECD). This participation includes the evaluation of all events (reactor, fuel facility, and licensed materials), except medical mis-administrations, reported to the NRC Operations Center for potential reporting using the INES. Reports are issued for all events rated as level 2 or higher, or as requested by another INES member nation.

Groundwater Tritium events that have been reported to NRC have all been evaluated as a Level "0", or no safety significance, we would not typically report these events.

The INES Manual calls for issuance of an INES rating to the IAEA for events rated at Level 0 or 1 if the event generated high international interest. Recently, in noting the significant levels of domestic media attention given to the tritium leakage into the ground at Vermont Yankee, staff detected little interest in this issue from the international regulatory community. Staff contacted the Nuclear Energy Agency in Europe to determine whether they had received any expression of media or public interest, and there has been none noted. Accordingly, these level "0" events have not met the criteria established for reporting events to the INES system.

Question 5: (to Chairman Jaczko on 3/9/10) Can you please address the tritium issue as it applies to license renewal for existing plants and for future plants?

Answer 5: As a routine part of our environmental site audit, the NRC reviews the applicant's Groundwater Protection Program. Staff's conclusions about tritium (or other radionuclides) are included in every license renewal Environmental Impact Statement, as we discuss (1) the potential for human health impacts, and (2) in the hydrology section for any potential impacts to groundwater quality. Future plants are required by regulation to describe how design features and procedures for operation will minimize the contamination of the facility and environment. This would include the potential for the release of tritium and other radionuclides into the environment.

Question 6: (to Chairman Jaczko on 3/9/10) In the recent media and Congressional blitz regarding tritium, you responded that the NRC was going to send inspectors to help find the leaks. This was unprecedented. Since when does the REGULATOR help the licensee do their job? The perceived message was that the tritium leak is so dangerous that the NRC regulator has to go help the licensee. Do you feel that the public understand that the tritium leak has very low safety significance? Or were your statements political posturing for the Vermont Senators and Congressman?

Answer 6: The NRC sent inspectors to VY in accordance with our normal processes to ensure that we understood the licensee's response to the rising levels of tritium in their monitoring wells. We did the same for Oyster Creek and some other plants.

Question 7: What has happened in consequence of the January letter from Congress to the GAO?

Answer 7: The GAO has not contacted the NRC regarding the Congressional request.

Question 8: Are industry and the NRC looking at tritium induced and accelerated corrosion?

Answer 8: Tritium is hydrogen that is chemically identical to normal hydrogen. Therefore, water that contains tritium is not different from a corrosion perspective from water that does not contain tritium. Tritium does not induce, accelerate or change the corrosivity of water.

Question 9: When does the NRC plan to begin their efforts at oversight (TI/audits) of buried piping programs?

Answer 9: The NRC has not yet established a schedule for issuing a temporary instruction of for performing audits of Buried Piping Integrity Initiative activities.

Question 10: The NRC has two responsibilities in its oversight and enforcement of a license condition: 1) safety, and 2) radiological control. Please explain why GDC 60 and GDC 64 are not violated by uncontrolled and unmonitored radioactive leaks from buried piping.

Answer 10: General Design Criterion 60, “Control of releases of radioactive materials to the environment,” states: “The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents…” Minor amounts of leakage from piping can be suitably controlled by isolating and repairing the affected piping.

General Design Criterion 64, “Monitoring radioactivity releases,” states: “Means shall be provided for monitoring… effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations….” Minor amounts of leakage have been identified from buried piping at a few plants. The leakage has been discovered by groundwater monitoring wells, and has been corrected. The groundwater monitoring wells provides means to monitor discharge paths and the plant environs.

Session Day and Time: Thursday, March 11, 2010, 8:30am – 10:30am

Session Number and Title: TH23 Emergency Preparedness Regulatory and Guidance Initiatives

Session Chair: Melvyn Leach, NSIR

Session Coordinator: Sara Mroz, NSIR, 301-415-1692, e-mail: Sara.Mroz@nrc.gov

Question 1: "When the new EP rule is implemented, including the requirement for license amendment for reduction in effectiveness, will implementing changes in E plans require license amendment? (i.e., will any changes likely be considered a reduction in effectiveness over previous E plans?)"

Answer 1: The NRC does not expect that changes to emergency plans to meet the requirements of the final rule would require prior NRC approval. Nonetheless, because licensees would be revising their emergency plans, each licensee would be required to evaluate its implementation of the final rule changes under 10 CFR 50.54(q) on a license-specific basis and, if required by § 50.54(q), submit the proposed changes for prior NRC approval.

Session Day and Time: Thursday, 8:30 a.m. – 10:30 a.m.

Session Number and Title: TH24 Increasing Interest in Small Modular Reactors

Session Chair: Stewart Magruder, Branch Chief, Advanced Reactors Program, NRC/NRO

Session Coordinator: John Smith, NRC/NRO, tel: (301) 415-0227, e-mail: John.Smith@nrc.gov

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

Session Day and Time: Thursday, 8:30 a.m. – 10:30 a.m.

Session Number and Title: TH25 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

All questions raised were answered during the session.

Session Day and Time: Thursday, March 11, 8:30- 10:30 a.m.

Session Number and Title: TH26 USNRC Safety Culture

Session Chair: Martin J. Virgilio, Deputy Executive Director for Materials, Waste, Research, State, Tribal, and Compliance Programs (DEDMRT)

Session Coordinator: Maria E. Schwartz, Office of Enforcement/Concerns Resolution Branch; 301-415-1888; e-mail: Maria.Schwartz@nrc.gov

Question 1: How do you pass safety culture values to the next generation?

Answer 1: The most effective means of transferring SC values to the next generation is through modeling the expression of those values in how day-to-day decisions are made and work is performed, augmented with both formal and informal teaching.

Question 2: What is the role foreseen for the Institute of Nuclear Power Operations (INPO) in the future safety culture assessment process?

Answer 2: NRC encourages the industry to take ownership of its safety culture initiatives and to continue to implement processes that results in a safer plant. The nuclear power industry is pilot testing the Nuclear Energy Institute (NEI) approach to safety culture assessment (NEI 09-07, *Fostering a Strong Nuclear Safety Culture*) at four plants, and the NRC is observing the industry's pilot activities. The industry assessment of nuclear safety culture incorporates the terminology of the Institute of Nuclear Power Operations (INPO) principles and attributes described in its *Principles for a Strong Nuclear Safety Culture*. While the lessons learned from the pilot will provide insights that will be considered during the implementation phase of the safety culture policy for nuclear power reactors, the NRC will retain independent oversight of licensee safety culture through the Reactor Oversight Process.

Question 3: If we as the leader feel confused by different definitions, assessments, measurements, etc., what do you think our employees would think? We need one approach to this important topic.

Answer 3: The NRC agrees with this comment. One of the goals of the NRC's February 2010 Safety Culture Workshop was to attempt to use common terminology as we develop a common definition of safety culture and a common set of traits/characteristics of a strong safety culture.

Question 4: You mentioned safety culture in security. Is the NRC concerned with the safety culture of licensee contractors and subcontractors? How is NRC addressing this?

Answer 4: The NRC is considering the inclusion of vendors in its policy statement on safety culture. It is anticipated that this will increase the awareness of safety culture in the ranks of contractors and subcontractors. Additionally, the quality assurance requirements of 10 CFR App B have a strong correlation to the safety culture objectives. These are required by regulation to be passed from licensees and applicants to their contractors and subcontractors.

Question 5: Are you results of the two industry pilot programs going to be incorporated into the policy statement?

Answer 5: The safety culture policy statement will provide a common definition and language of safety culture that applies equally to the breadth of NRC licensees. Consequently, the policy will necessarily be written at a high level. A strategy for implementing the safety culture policy for each licensee type will be developed. The nuclear power industry is pilot testing the Nuclear Energy Institute (NEI) approach to safety culture assessment (NEI 09-07) at four plants, and the NRC is observing the industry's pilot activities. While the lessons learned from the pilot will provide insights that will be considered during the implementation phase of the safety culture policy for nuclear power reactors, it is unlikely that the results will be directly incorporated into the policy statement.

Question 6: How do you distinguish the current process difference in safety culture between a single unit site that has 1 refueling outage once every 18 months and a 3 unit site with a much larger staff and 2 refueling outages per year when the assessment is based on a count and still maintain the process open and transparent?

Answer 6: The NRC's process for identifying cross-cutting aspects and substantive cross-cutting issues (SCCIs) is described in NRC Inspection Manual Chapter 0305. This process does not differentiate between single- and multi-unit sites. The number of cross-cutting aspects identified at a site is only one consideration of the SCCI decision-making process. Another important component of this process is the NRC's confidence in the licensee's scope of efforts or progress in addressing the cross-cutting theme(s). Although a licensee may have multiple findings with the cross-cutting aspect, the NRC may not identify an SCCI if the licensee is taking adequate corrective actions to address the theme. As such, the fact that the staff will identify an SCCI is in large part a function of its confidence in the licensee's corrective actions and is independent of the number of reactor units at a given site.

Question 7: If a plant is found to have an "unacceptable safety culture," what is the expected NRC response and are there expected actions that the licensee should take? Related question: How will NRC take enforcement action/ensure compliance to correct a weak safety culture?

Answer 7: The NRC does not make determinations that a licensee's safety culture is "unacceptable." However, routine (baseline) NRC inspections can provide insights into areas important to safety culture, and the NRC may focus inspection effort on reviewing licensees' corrective actions for cross-cutting issues. The NRC also reviews safety culture components as part of its supplemental inspection program and may request licensees to perform safety culture assessments in response to degrading performance that causes the licensee to be in the Degraded Cornerstone or Multiple/Degraded Cornerstone Columns of the ROP Action Matrix. The Reactor Oversight Process has many flexibilities available to the staff and includes an action matrix that describes the levels of NRC oversight and inspection based on plant performance. For example, if a licensee moves into the multiple/repetitive degraded cornerstone column (known as column 4) of the action matrix, the NRC will initiate a supplemental inspection program to investigate the extent of safety culture problems. Further, the NRC may request that the licensee conduct an assessment of its safety culture and develop corrective actions which will be assessed during future NRC inspections. As it deems appropriate, the

NRC itself may also conduct in-depth inspections of safety culture at "column 4" sites as part of its supplemental inspection effort. The NRC will continue its increased oversight activities until the licensee can demonstrate that it has improved its processes and performance such that the NRC has renewed confidence in the licensee's ability to identify and resolve its performance issues.

Question 8: Everything about the NRC process for understanding safety culture is transparent except the part about "confidence". How does the NRC measure their confidence in the management team? Is the measure for measuring it quantifiable?

Answer 8: The NRC's process for identifying a substantive cross-cutting issue (SCCI) is described in Section 13.02 of NRC Inspection Manual Chapter (IMC) 0305. An SCCI exists if (1) a cross-cutting theme(s) in the cross-cutting areas exists and (2) the NRC staff has a concern with the licensee's scope of efforts or progress in addressing the cross-cutting theme(s).

In evaluating whether the second criterion (i.e., the "confidence criterion") is met, IMC 0305 provides guidance to the staff to consider if any of the following situations exists: (1) the licensee had not identified or recognized the cross-cutting theme(s) affected other areas and had not taken actions to address the theme(s); (2) the licensee recognized the cross-cutting theme(s) affected other areas but failed to schedule or take appropriate corrective action; (3) the licensee recognized the cross-cutting theme(s) affected other areas but did not implement timely corrective actions commensurate with the significance of the issue(s); and (4) the licensee has implemented a range of actions to address the crosscutting theme(s); however, these actions have not yet proven effective in substantially mitigating the cross-cutting theme(s) even though a reasonable duration of time has passed.

The "confidence criterion" is based on the effectiveness of the licensee's corrective actions to address the cross-cutting theme. Inspection Procedure (IP) 71152, "Problem Identification and Resolution," instructs inspectors to review a licensee's corrective actions for cross-cutting themes and SCCIs. This IP lists attributes that inspectors consider when determining the effectiveness of licensees' corrective actions.

Question 9: Related to the Reactor Oversight Process (ROP), what is your opinion about the "pro-activity" of the KPI's of the reactor oversight? Are the precursors included?

Answer 9: The objectives of the Reactor Oversight Process (ROP) are to provide tools for inspecting and assessing licensee performance in a manner that is more risk-informed, objective, predictable, and understandable than the previous oversight process. The associated philosophy of the ROP involves the use of indicative measures of performance (e.g., performance indicators (PIs) and inspection findings) rather than predictive measures of performance. However, the staff continues to explore the potential use of leading indicators of performance in response to stakeholder feedback.

PIs, together with risk-informed baseline inspections, are intended to provide a broad sample of data to assess licensee performance in the risk-significant areas. Objective performance evaluation thresholds were intended to help determine the level of regulatory engagement appropriate to licensee performance.

The PI thresholds were selected to be risk-informed to the extent practical, but they also accommodated defense-in-depth strategies, regulatory requirements, and safety analyses. The thresholds were established so that sufficient margin existed between nominal performance bands to allow for licensee initiatives to correct performance problems before reaching escalated regulatory involvement.

Question 10: How will NRC identify violation of intent of safety culture requirements in its allegations program? That is, deal with those who "game" the Reactor Oversight Process?

Answer 10: The agency's Allegation Program administers the assessment of safety concerns received by the NRC from outside sources. Although, NRC encourages workers at regulated nuclear facilities to take safety concerns to their own management first, workers can bring safety concerns, including those associated with alleged weaknesses in the safety culture, directly to the NRC at any time. It is the agency=s responsibility to independently evaluate and respond to those concerns in a timely manner.

Session Day and Time: Thursday, 8:30 a.m. – 10:30 a.m.

Session Number and Title: TH27 Thermal Hydraulics and Severe Accident Code Development and Application

Session Chair: Jennifer Uhle, Director, Division of Systems Analysis, NRC/RES

Session Coordinator: Antony Calvo, NRC/RES, tel: (301) 251-7677,
e-mail: Antony.Calvo@nrc.gov

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

Session Day and Time: Thursday, 11:00 a.m. – 1:00 p.m.

Session Number and Title: TH28 Advisory Committee on Reactor Safeguards – The Latest Chapter

Session Chair: Edwin Hackett, Executive Director, NRC/ACRS

Session Coordinator: Kendra Freeland, NRC/ACRS, tel: (301) 415-7998,
e-mail: Kendra.Freeland@nrc.gov

All questions raised were answered during the session.

Session Day and Time: Thursday, 11:00 a.m. – 1:00 p.m.

Session Number and Title: TH29 Construction Inspection Program

Session Co-Chairs: Glenn Tracy, Director, Division of Construction, Inspection, and Operational Programs, NRC/NRO, and Richard Croteau, Director, Division of Construction Projects, NRC/Region II

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

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

Session Day and Time: Thursday, 11:00 a.m. – 1:00 p.m.

Session Number and Title: TH30 Fire Protection

Session Chair: Sunil Weerakkody, Deputy Director for Fire Protection, Division of Risk Assessment, NRC/NRR

Session Coordinator: Daniele Oudinot, NRC/NRR, tel: (301) 415-4064, e-mail: Daniele.Oudinot@nrc.gov

All questions raised were answered during the session.

Session Day and Time: Thursday, March 11, 2010, 11:00 am – 1:00 pm

Session Number and Title: TH31, "Potential Changes to the U.S. Low Level Waste Regulatory Framework"

Session Chair: Patrice M. Bubar, FSME

Session Coordinator: James Kennedy, FSME, 301-415-6668, e-mail: James.Kennedy@nrc.gov

Question 1: If the intruder scenario were no longer limiting for some radionuclides, could those radionculides become limiting in other pathways, e.g., the impact on the groundwater pathway?

Answer 1: Hypothetically, the answer could be yes. However, as part of the staff's future rulemaking efforts, they intend to perform technical analyses to identify specific radionuclides that might bound both the intruder, concentration-based scenario as well as a activity-based, drinking water scenario.

Question 2: Under a site-specific analysis, if a waste generator assembled a LLW package that met disposal site “A” criteria but later found that they could not ship LLW to that site, describe the implications for waste re-packaging, handling exposures to workers, and the potential [sic] for orphaned waste.

Answer 2: We understand this question to mean that if a waste package fails to meet the independent acceptance criteria of some regional compact as opposed to the waste acceptance criteria (WAC) of the disposal facility itself, there is the potential for additional radiation exposures due to the need to repackage to waste to meet the compacts’ independent acceptance criteria.

Under this scenario, there would likely be additional radiation exposures. Consequently, to minimize all exposures, it is important for waste generators, in the first instance, to be versed in both the regional compacts’ independent acceptance criteria as well as the WAC for the disposal facility in question.

As regards the potential for orphaned waste, we believe that if the generator complies with both the regional compacts’ independent acceptance criteria and the disposal facility WAC, this scenario can be avoided.

Question 3: As regards risk-informing the LLW classification scheme, what parameters will be used and how will they be measured?

Answer 3: The adoption of a risk-informed LLW classification scheme implies that appropriate regulatory measures will be used to reduce and/or mitigate radiation exposures to workers as well as general members of the public. To this end, standard health physics techniques rely on a reasonable set of exposure scenarios that could be used to estimate potential radiation exposures to workers and the public. Imbedded in those techniques are a combination of both deterministic and probabilistic models and parameters derived from a review of the scientific literature that has been peer-reviewed.

In reference to those parameters that might subject to measurement ‘per se’, composition of the waste spectrum that could potentially go into a LLW disposal facility is one example. To determine the types and kinds of materials defining the current LLW spectrum and subject to future regulation, it is possible that the staff will undertake a survey to canvas potential generators known to be producing LLW. Using this survey information, the staff can determine which radionculides and in what concentrations are dominating postulated doses. In this fashion, the regulatory framework should be risk-informed based on verified knowledge concerning the known radiation hazards.

Question 4: Security and safety are intertwined. What consideration is or will be given to security in the development of a risk-informed framework for LLW?

Answer 4: NRC’s current regulation for LLW at Part 61 provides adequate assurances against unauthorized access to a LLW disposal facility. These measures (regulations) would not be expected to change under any future risk-informed LLW rulemaking as they are standard provisions for all NRC licensees. As is the case generally with NRC’s regulations pertaining to materials, the foremost goal is the protection of public health and safety. This goal will not change under a risk-informed regulatory paradigm. NRC’s waste classification system is principally designed to ensure the safety of an inadvertent intruder into a waste disposal facility.

Security of disposed waste has not been an issue in the past, but the staff will consider if any additional security considerations are warranted in light of the increased focus on security since 9/11.

Question 5: if the waste from the Savannah River site is not ultimately disposed of at the Energy Solutions Class A disposal facility in Clive, Utah, where will DOE send it to for disposal? Can it go to the Waste Isolation Pilot Plant (WIPP) facility in Carlsbad, New Mexico?"

Answer 5: The question is interpreted to apply to depleted uranium currently at the Savannah River site. DOE has not made any decision at this time on where the waste will be disposed, if it is not disposed at EnergySolutions at Clive. The waste cannot go to the WIPP facility because the waste is not considered to be defense transuranic radioactive waste.

Session Day and Time: Thursday March 11, 2010, 11:00 a.m. – 1:00 p.m.

Session Number and Title: TH32 Spent Nuclear Fuel Criticality Analysis Issues

Session Chair: Sher Bahadur, Deputy Director, Division of Safety Systems, NRC/NRR

Session Coordinator: Kent Wood, Reactor Systems Engineer, Division of Safety Systems, NRC/NRR, (301) 415-4120, e-mail: Kent.Wood@nrc.gov

Question 1: With regard to "reduced margins" what is driving the need/desire for these reduced margins?

Answer 1: The primary driver behind the spent fuel pool licensing amendment requests the NRC had received over the past four years is a need to increase or maintain SFP storage capacity.

Question 2: You [Presenter Dr. Lancaster] stated a preliminary result of 2% for total uncertainty (Δk_d). Is that stated as a total Δk or as a percentage of burnup related Δk. Please explain [the] difference related to historical application of "5% of BU [burnup] related credit".

Answer 2: Response provided by Dr. Lancaster: The 2% in k estimate was for 20 GWD/MTU. The estimated Δk_d is a constant 0.004 in k and not a function of burnup. The historical application is 5% of the delta k of depletion and clearly increases with burnup. The real uncertainty related to change in k with burnup would have to be 0 at 0 GWD/MTU. We see no trend in our ability to predict k as a function of burnup (in startup ppm or predicted cycle length - power distribution work is ongoing but no trend has been observed yet). The 0.004 is simply the uncertainty in our ability to predict k. I have conservatively assumed that all of the uncertainty in our prediction of k is due to the uncertainty in the change in k with burnup. In fact, from the analysis of critical experiments the uncertainty in k for fresh fuel is about 0.006. If we could subtract this non-burnup uncertainty, it leads to the conclusion that the uncertainty in the change in k due to burnup is insignificant compared to the uncertainty in the predicting k for fresh fuel.

Question 3: Is there any value in pursuing "integral spent fuel critical experiments." Is there any initiative to perform such experiments at a shutdown reactor site with an operating license prior to decommissioning [Questioner made reference the recent Vermont legislature vote with respect to Vermont Yankee.].

Answer 3: Yes, there is value in pursuing critical experiments in which spent nuclear fuel is appropriately represented. Some work has been done in this regard. The experiments in NUREG CR-6979 Evaluation of the French Haut Taux de Combustion (HTC) Critical Experiment Data, are examples of that work. There is an international effort through OECD/NEA to identify what experiments would provide the most benefit. The NRC attended the most recent workshop associated with this effort. There is no initiative to conduct experiments in a NRC licensed SFP. To the best of the NRC's knowledge there is no initiative to conduct experiments in an internationally licensed SFP. Such experiments would be impractical.

Session Day and Time: Thursday, 11:00 a.m. – 1:00 p.m.

Session Number and Title: TH33 Trends in Shutdown Events

Session Chair: Lois James, Branch Chief, Division of Risk Assessment, NRC/NRR

Session Coordinator: Fernando Ferrante, NRC/NRR, tel: (301) 415-8385, e-mail: Fernando.Ferrante@nrc.gov

All questions raised were answered during the session.

Session Day and Time: Thursday, 2:30 p.m. – 4:30 p.m.

Session Number and Title: TH34 Enhancements to the USNRC's Force-on-Force Inspection Program

Session Chair: Barry Westreich, Deputy Director for Security Oversight, Division of Security Operations, NRC/NSIR

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

All questions raised were answered during the session.

Session Day and Time: Thursday, 2:30 p.m. – 4:30 p.m.

Session Number and Title: TH35 External Flood and Extreme Precipitation Hazard Analysis for Nuclear Plant Safety

Session Chair: Thomas Nicholson, Senior Technical Advisor for Radionuclide Transport, Division of Risk Analysis, NRC/RES

Session Coordinator: Joseph Kanney, NRC/RES, tel: (301) 251-7600,
e-mail: Joseph.Kanney@nrc.gov

Question 1: Since the age of the earth is less than 1E+10 years and Reactor Safety Goals are evaluated for events with recurrence intervals greater than 1E+5 years, why are unlimited, unbounded surge calculated?

Answer 1. We are not attempting to calculate an unbounded surge estimate. Quite the opposite, we are attempting to derive a reasonable (physically based) upper bound estimate. The reason we do this do this is that our experience with extreme events is so limited that we cannot reliably assign return periods in the same way that we can with hydrological phenomena of smaller magnitudes or in the way that we estimate failure probabilities of engineered systems such as reactors.

Session Day and Time: Thursday, 2:30 p.m. – 4:30 p.m.

Session Number and Title: TH36 GSI-191 Looking Back and Going Forward

Session Chair: Michael Scott, Branch Chief, Division of Safety Systems, NRC/NRR

Session Coordinator: Michael Scott, NRC/NRR, tel: (301) 415-0565,
e-mail: Michael.Scott@nrc.gov

All questions raised were answered during the session.

Session Day and Time: Thursday, March 11, 2:30 p.m. - 4:30 p.m.

Session Number and Title: TH37 Next Generation Nuclear Plant (NGNP) Research

Session Chair: John R. Jolicoeur, RES/DSA/NARB

Session Coordinator: Michael B. Rubin, RES/DSA/NARB, 301-251-7549,
e-mail: MichaelB.Rubin@nrc.gov

All questions raised were answered during the session.

Session Day and Time: Thursday, 2:30 p.m. – 4:30 p.m.

Session Number and Title: TH38 Reactor Oversight Process – Retrospective and Future

Session Chair: Frederick Brown, Director, Division of Inspection and Regional Support, NRC/NRR

Session Coordinator: Leslie Fields, NRC/NRR, tel: (301) 415-1186, e-mail: Leslie.Fields@nrc.gov

All questions raised were answered during the session.

Session Day and Time: Thursday, March 11, 2010, 2:30 pm – 4:30 pm

Session Number and Title: TH39 Risk Communication – Principles, Tools, and Techniques

Session Chair: Brian McDermott, NSIR

Session Coordinator: Sara Mroz, NSIR, 301-415-1692, e-mail: Sara.Mroz@nrc.gov

All questions raised were answered during the session.

Session Day and Time: Thursday, 2:30 p.m. – 4:30 p.m.

Session Number and Title: TH40 Risk-Informed Licensing Actions: Recent Experience and New Initiatives

Session Chair: Michael Markley, Branch Chief, Division of Operating Reactor Licensing, NRC/NRR

Session Coordinator: Michael Markley, NRC/NRR, tel: (301) 415-5723, e-mail: Michael.Markley@nrc.gov

All questions raised were answered during the session.