NRC 20th Annual Regulatory Information Conference (RIC)

IOUs and Action Items

March 11 – 13, 2008

Track 1	General Plenary Sessions	
Track 2	Operating Reactors	
Track 3	Reactor Research	
Track 4	New Reactors	
Track 5	Security, Emergency Preparedness, Fuel Cycle	
Track 6	Regional Breakout	

Below are questions that were not addressed by the chair or panel members during technical sessions. Answers will be posted as they are provided. For more information, please contact the designated session coordinator.

Tuesday, March 11, 2008

4:00 pm – 5:30 pm Salon F-H		**Emerging Issues: Materials and Mechanical** Track 2 – Operating Reactors All questions answered onsite. **Session Chair:** Michele Evans, NRC/NRR **Session POC:** Leslie Miller, NRC/NRR, tel: (301) 415-1037 e-mail: LSM2@nrc.gov
4:00 pm – 5:30 pm Salon D		**Environmental Reviews for New Reactors: Looking Back and Looking Forward** Track 4 – New Reactors **Question 1:** Requirements related to EMS and AMS apply to activities carried out by federal agencies. It seems like they do not apply to nuclear plant owners who are not federal agencies. How do you see these fitting into a regulator/licensee environment? **Question 2:** How broad of a scope should be considered when conducting cumulative effects analysis? Time frame? Previous impacts? **Question 3:** Boling, given NEPA and CEQ requirements, and your familiarity with other Agency practices, what is your perspective on NRC's potential new interpretation that would require ERS to segregate non-NRC regulated activities? **Question 4:** How many NEPA cases have involved NRC? **Question 5:** When predicting cumulative efforts, does consideration have to be given to

		possible accidents? **Session Chair:** Jim Lyons, NRC/NRO **Session POC:** Tamsen Dozier, NRC/NRO, tel: (301) 415-2272 e-mail: TSD2@nrc.gov
4:00 pm – 5:30 pm Brookside		**Emergency Preparedness and Incident Response** Track 5 – Nuclear Security, Emergency Preparedness, Fuel Cycle All questions answered onsite. **Session Chair:** Mel Leach, NRC/NSIR **Session POC:** Ned Wright, NRC/NSIR, tel: (301) 415-5563 e-mail: NXW1@nrc.gov
4:00 pm – 5:30 pm Salon E		**Getting Ahead of Performance Issues** Track 2 – Operating Reactors **Question 1:** In the area of OpE, should the licensees be held accountable for equipment issues from outside the nuclear industry for a piece of equipment that is installed in our plants? **Answer 1:** Licensees benefit from applying OpE from all sources. Licensees should be aware of what equipment is installed in their plant and should obtain available OpE for that equipment consistent with the provisions of Appendix B and the Maintenance Rule (10 CFR 50.65). **Session Chair:** Fred Brown, NRC/NRR **Session POC:** Steve Vaughn, NRC/NRR, tel: (301) 415-3640 e-mail: SJV1@nrc.gov
4:00 pm – 5:30 pm Salon A-C		**State of the Art Reactor Consequence Analysis (SOARCA)** Track 3 – Reactor Research **Question 1:** What will the new models and results from this study be used for? Were these purposes known and used to direct the project scope, level of detail, and analysis methods? **Question 2:** What impacts will the Feb. 25 ACRS letter to the NRC Chairman on SOARCA have on the project? **Question 3:** Is there an experimental basis for determining operability of severe accident mitigation equipment, considering the extreme environments? **Question 4:** You mentioned that we are including 1% of the sequences of importance to safety, so we are not using an exclusive approach. This was not clear, please clarify. **Session Chair:** Farouk Eltawila, NRC/RES **Session POC:** Alison Rivera, NRC/RES, tel: (301) 415-5059 e-mail: ALD2@nrc.gov

Wednesday, March 12, 2008

11:00 am – 12:30 pm Salon D		**Construction Inspection Program** Track 4 – New Reactors **Question 1:** Dolan, in your opinion, will the reactor vendors have detailed design complete when construction begins? Tracy, what actions can/will NRC take if detailed

		case circuit breakers. **Answer 1:** The NRC staff recently took action to raise awareness of the potential for suspect/counterfeit parts and the potential for such parts to enter the nuclear industry supply chains by issuing Information Notice (IN) 2008-04, "Counterfeit Parts Supplied to Nuclear Power Plants." The Square D molded case circuit breaker issue was specifically discussed. Prior to issuing the IN, the NRC staff made a presentation, "NRC Perspectives on Dedication and Counterfeit Products," to industry representatives involved in procuring components for nuclear power plants at a meeting of the Electric Power Research Institute. The presentation discussed the Square D molded case circuit breakers. Additionally, the staff discussed some past issues and recommended reviewing some past generic communications that had documented cases of suspect/counterfeit parts. The presentation is publically available in ADAMS (accession number ML080950487). The staff also discussed the issue of counterfeit parts with the Nuclear Procurement Issues Committee. **Session Chair:** Patrick Hiland, NRC/NRR **Session POC:** Kerby Scales, NRC/NRR, tel: (301) 415-1369 e-mail: KVS1@nrc.gov
2:00 pm – 3:30 pm Salon D		**International Activities on New Reactors** Track 4 – New Reactors **Question 1:** Considering that it is typical for vendors to interact with and support regulators reviewing their designs, what processes are being implemented to facilitate vendor support for the MDEP design working groups? **Answer 1:** Membership in MDEP is limited to national regulatory authorities. The industry does have parallel efforts such as the World Nuclear Association program on international standardization, and the cooperative discussions among Codes and Standards organizations. MDEP will communicate and cooperate with such organizations, in an open manner, where appropriate. Vendors presenting their reactor designs to multiple regulators can best support the MDEP by ensuring that they communicate openly and consistently with all regulators. **Question 2:** Understanding that there is a need to firmly establish MDEP's processes for reactors, what is the likelihood of eventually extending MDEP to include fuel cycle facilities (e.g., recycling plants)? **Answer 2:** Neither the NRC nor the MDEP currently have plans to extend the MDEP program to fuel cycle facilities. **Question 3:** Is training and licensing of operators consistent from country to country; design to design? **Answer 3:** The MDEP is design-centered program, and was not intended to address operator licensing and training. As such, no data has been gathered under the MDEP program on operator licensing or training. **Session Chair:** Gary Holahan, NRC/NRO **Session POC:** Robert Elliott, NRC/NRO, tel: (301) 415-1397 e-mail: RBE@nrc.gov
2:00 pm – 3:30 pm Lower Level Entrance Marinelli & Executive Blvd		**Incident Response Experience** Track 5 – Nuclear Security, Emergency Preparedness, Fuel Cycle All questions answered onsite.

expect vendors to coordinate with the NRC for scheduling of their activities?

Answered 8: The current plan for scheduling vendor inspections does not tie directly to verification of ITAAC. The approach of the vendor inspection program for new reactors is to assess the overall technical adequacy and quality of parts and services provided by the vendors. Scheduling of vendor inspections, therefore will typically be based on factors beyond ITAAC. These factors include, for example, industry experience with vendors, operating or construction experience, the scope of parts and services provided by specific vendors, and NRC experience with vendors. There may also be instances where a vendor inspection is conducted specifically to verify an element of an ITAAC. These will be conducted on a case-by-case basis and will be coordinated with the licensee constructing the facility to assure the NRC witnesses the activities needed to support ITAAC closure verification. For these limited cases, the NRC will coordinate schedules, through the licensee, with the vendor manufacturing or fabricating the parts or components.

Question 9: In QA/QC, what efforts are being taken to eliminate or minimize duplication of efforts, inspections, and audits between INPO, ANI, utilities, and the NRC?

Answer 9: None. These efforts are intended to be independent of one another.

Question 10: Can you provide some details of what types of failures may result in escalated enforcement action?

Answer 10: Escalated enforcement actions are any Severity Level (SL) I, II, or III violations. Examples of various escalated enforcement actions were discussed during the December 18th, 2007 public meeting. Examples of failures for each severity level include:

Severity Level I:
A. A violation involving multiple structures, systems, or components that are completed in a manner such that they would not have satisfied their intended safety purpose.

Severity Level II:
A. A significant breakdown in the Quality Assurance (QA) program as exemplified by multiple deficiencies in construction QA related to more than one work activity (e.g., structural, piping, electrical, foundations). These deficiencies involve the licensee's failure to provide adequate oversight or take prompt corrective action and involve multiple examples of deficient construction or construction of unknown quality due to inadequate program implementation.
B. A structure, system, or component that is completed in such a manner that it could have an adverse impact on the safety of operations.
C. Widespread significant failures of licensee QA program, involving ineffective oversight, failures of multiple barriers in the licensee's programs, and corrective action measures, affecting multiple structures systems, or components.

Severity Level III:
A. A breakdown in a licensee QA program for construction related to a single work activity (e.g., structural, piping, electrical, foundations). This significant deficiency involves the licensee's failure to consistently provide adequate oversight or take prompt corrective action, and involves multiple examples of deficient construction or construction of unknown quality due to inadequate program implementation.
B. A failure to confirm the design safety requirements of a structure, system, or component as a result of inadequate pre-operational test program implementation.
C. Ineffective corrective actions, resulting in multiple examples of recurring significant deficiencies associated with a single construction activity.

Question 11: Will the electronic construction inspection documentation system accommodate security inspections (i.e. non-public information)? Will security ITAAC closure information (inspections, etc.) be public?

Answer 11: The details of the security inspection program have not been fully developed at this time. However, the ITAAC that have a "security" connection are already public. Inspecting the security hardware that is already "public" (because of the ITAAC or other DCD or FSAR information) will likely be public in the NRC inspection reports and in our CIPIMS data provided the inspection does not reveal site specific details that would otherwise be considered sensitive.

For the security program information, that would be safeguards material, this is more likely part of IMC-2504 for operational programs. Our current plan requires this information to be handled similar to what we did for the same type of inspections (part of IMC-2513, Appendix B) for the Part 50 plants. The NRC inspection reports that cover safeguards material will have a paragraph title indicating what was inspected and then underneath that title will be a standard statement stating "this is safeguards information and not available for public review".

Question 12: Is the public involved in the ITAAC process? If so, how do they play? How is the CAP process associated with construction seems to be the handling of field/construction changes usually found by inspection and or construction oversight now being captured under a CAP process?

Answer 12: Documents related to the ITAAC closure process such as the licensee's ITAAC notification letters required by 10 CFR 52.99, NRC ITAAC inspection reports, and the NRC's Federal Register Notices of ITAAC completion, will be publicly available. Under 10 CFR 52.103(a), the NRC will publish a notice of intended operation in the Federal Register not less than 180 days before the date scheduled for initial loading of fuel into a reactor by a licensee that has been issued a combined license. The notice will provide that any person whose interest may be affected by operation of the plant may, within 60 days, request that the Commission hold a hearing on whether the facility as constructed complies, or on completion will comply, with the acceptance criteria in the combined license In addition to the contention and standing requirements of 10 CFR 2.309, such a request for hearing must show, prima facie, that one or more of the acceptance criteria of the ITAAC in the combined license have not been, or will not be met. The request must also identify the specific operational consequences of nonconformance with the ITAAC that would be contrary to providing reasonable assurance of adequate protection of the public health and safety.

Regarding the relationship between the CAP process and the process for resolving field/construction changes, these processes are separate and distinct. An item entered into the CAP may also need to be entered into the engineering change process for the ultimate resolution, but not all items that require change will require a CAP entry. Changes are expected during the course of construction and licensee procedures should discuss the required work flow process. Items that are outside of the licensee procedures would require a CAP entry because they represent a breakdown of the licensees process.

Question 13: How will ITAAC inspections by NRC be fundamentally different for new COL Part 52 construction compared with the 1970s and 1980s Part 50 plant construction?

Answer 13: The actual NRC inspections of "processes" and "programs" that will be performed are very similar to that which was implemented for Part 50 facilities. The part 50 inspection process relied on inspections to verify that design, implementation, and programs were built or established to support a conclusion that the facility should be licensed to operate. In the case of Part 52, the facility is licensed, and the ITAAC inspection process is designed to confirm that the facility was built in accordance with

		the licensee prior to operation. ITAAC inspections are one piece of a larger review. ITAAC inspections along with the certified design review, the COL review, and 2501/2502/2504 inspections will encompass all activities performed using the part 52 process. **Question 14:** Grier, you indicated in an answer to a question, that when a significant finding occurs you notify NUPIC systems and members. How does the global nuclear power industry find out? Through NRC? Through the grapevine? **Answer 14:** Our major focus is to share information among our membership. If the issue is significant enough and could require 10CFR21 notification, the condition would be reported by either the supplier or the member and would be available through the NRC. In some cases, it may also be reported through INPO Operating Experience. We are currently evaluating how we can share this supplier performance information with some of the major suppliers who will be procuring materials and services for the new plants. **Question 15:** Are there plans to address suppliers that work to ISO-9001 only meeting 10CFR52 and 10CFR50 App B requirements? **Answer 15:** NUPIC: We currently audit some ISO 9001 supplier quality programs to the requirements of 10CFR50 Appendix B. Suppliers who have taken the necessary steps to bridge the gaps between ISO 9001 and 10CFR50 Appendix B are usually found to be acceptable. We also perform commercial grade item surveys to assess supplier control of critical characteristics. **Question 16:** In the 1980s and 1990s, the industry and NUPIC did a good job of reviewing vendor QA programs and putting vendors on approved supplier lists. However, the reviews often did not detect significant product noncompliance and fraudulent parts. What has changed in the industry's approach to address these issues? **Answer 16:** We now have a better communication network when there are potential performance issues with suppliers. For Duke, supplier audits provide us with an indicator of potentially broader concerns with a supplier and we may provide additional oversight for products being procured. In some cases we have scheduled joint surveillance activities through NUPIC. We are also evaluating our audit process for enhancements in these areas. Information is also available through other industries (FAA, DOE) that we are reviewing for possible incorporation into our process. **Question 17:** Has consideration been given to placing personnel in supplier facilities to help establish a nuclear safety culture quality mindset? Audits may be too late to catch fraudulence. **Answer 17:** Yes, consideration has been given to the placement of resident inspectors at certain types of fabrication facilities (e.g., facilities where modules will be constructed); but not likely at individual component vendors. The supplier inspections are accomplished as part of the NRC Vendor Inspection Program. This was successfully implemented during the Part 50 plant construction and will continue to be implemented for Part 52 plant construction. **Question 18:** Our company has received a NIAC audit, but not a NUPIC audit. We frequently get asked to supply services to utilities directly. Do you think a NIAC audit will ever suffice for a NUPIC audit? **Answer 18:** Although I can't speak for NIAC, it is NUPIC'shttp://www.nrc.gov/public-involve/conference-symposia/ric/past/2008/index.html desire to work with NIAC. We feel we have a lot to offer that NIAC may want to incorporate into their audit process. One idea that surfaced recently was to offer NUPIC auditor training to NIAC supplier auditors.

Question 19: Does NUPIC anticipate any significant challenges from (a) the transition of many members from “45.2” programs to NQA-1 programs, (b) provisions of the Pt 52 process, including timing of license vs. Construction, and/or (c) the increased use of contractor staff for what some used to do in-house?

Answer 19: NUPIC member auditors are currently familiar with NQA-1 because many suppliers are using it as their program basis. The other issue that could have some impact is the auditor qualifications, but I don’t believe this will be a significant issue. We are also planning to provide standardized training for supplier auditors which we may also make available to suppliers. NUPIC currently has provisions for using contract staff and will continue to monitor and evaluate this practice as new plant construction moves forward.

Question 20: What special challenges does the panel see with the emphasis on modular construction, especially QA/QC and inspections?

Answer 20: Modular construction techniques will pose a challenge since their effective implementation will demand earlier and closer integration of efforts by designers and modular constructors. The NRC staff anticipates that U.S. reactor designers and Architect-Engineers will continue to be challenged by an extended global supply-chain. Similarly, QA/QC and inspections for modular construction will also require earlier and closer integration of efforts by licensees and NRC.

Question 21: To what extent has the NRC integrated its inspection program with the LWA process?

Answer 21: A Limited Work Authorization (LWA) is a license that authorizes only certain specific construction activities. For this reason, the timing of these inspection activities may change but the inspections that will be performed remain the same as those performed if an LWA was not issued. NRC has focused the effort to complete and issue inspection procedures on those that may be required to inspect LWA activities. These procedures were issued in October 2007.

Question 22: One of your three goals was to have design complete prior to construction. While a laudable goal, designs have not been complete, but construction has started. Specifically, COLAs are still under review but long lead times are being built now. How is NRC going to address inspection follow-up as design completes? Designs will only be complete to the maximum extent practical and some construction will continue in parallel with completing design. What mechanisms will the NRC rely on for configuration control as applicants/licensees complete design?

Answer 22: The NRC has identified the level of design detail required for design certification application in 10 CFR 52.47. The NRC expects at least this same level of detail in a combined license (COL) application. Construction can begin once the COL is issued. The NRC construction inspection process will evaluate the design implementation through the construction inspection program. Components that are already under procurement will be inspected through our vendor inspection program or as part of the site inspection program for material receipt. At the end of construction the facility must meet the design requirements specified in the license. Additionally, licensees must meet Criterion III of appendix B to 10 CFR Part 50, for Design Control, which includes requirements for the proper handling of design changes.

Question 23: Does NRC foresee reopening ITAAC already closed out and accepted by the NRC as closed? Is closed out ITAAC above allowed further hearing?

Answer 23: The NRC will not reopen ITAAC that have been verified to be closed unless the NRC receives information that would lead the NRC to conclude that the ITAAC may no longer be met. Closed ITAAC are not subject to hearing unless a contention brought

	by a party meets the contention and standing requirements of 10 CFR 2.309 as well as the additional standards outlined in 10 CFR 52.103(b). These additional standards are (1) that the contention shows prima facie that one or more of the acceptance criteria of the ITAAC in the combined license have not been, or will not be met, and (2) that the contention identifies the specific operational consequences of nonconformance with the ITAAC that would be contrary to providing reasonable assurance of adequate protection of the public health and safety. **Question 24:** Has the NRC considered the use of statistical process control (SPC) ITAACs sampling to select ITAACs? **Answer 24:** The NRC's sampling methodology is not a "statistical" process. The NRC selects ITAAC based on their "value of inspection" using a "smart" sampling methodology that has four attributes - safety significance, propensity for errors, opportunity to verify by other means, and construction and testing experience. **Session Chair:** Glenn Tracy, NRC/NRO and Loren Plisco, NRC/R-II **Session POC:** Roger Rihm, NRC/NRO, tel: (301) 415-7807 e-mail: RXR3@nrc.gov
11:00 am – 12:30 pm Salon E	**Lessons Learned from International Operating Experience** Track 2 – Operating Reactors **Question 1:** What can the NRC do to be more timely in issuance of generic communications? **Question 2:** How do you resource an operating experience team at the regulatory body? What would be a minimum compliment? Do you believe that the INES and IRS systems meet their purpose? Does the NRC use them? **Session Chair:** Mary Jane Ross-Lee, NRC/NRR **Session POC:** Greg Bowman, NRC/NRR, tel: (301) 415-2939 e-mail: GTB1@nrc.gov
11:00 am – 12:30 pm Lower Level Entrance Marinelli & Executive Blvd	**Incident Response Experience** Track 5 – Nuclear Security, Emergency Preparedness, Fuel Cycle All questions answered onsite. **Session Chair:** Brian McDermott, NRC/NSIR **Session POC:** Janelle Jessie, NRC/NSIR, tel: (301) 415-6775 e-mail: JRB6@nrc.gov
11:00 am – 12:30 pm Brookside	**Aging and Life Beyond 60: The Next License Renewal Period(s)** Track 3 – Reactor Research All questions answered onsite. **Session Chair:** Jennifer Uhle, NRC/RES **Session POC:** Gene Carpenter, NRC/RES, tel: (301) 415-7333 e-mail: CEC@nrc.gov
11:00 am – 12:30 pm Salon F-H	**Emerging Issues: Electrical –** **Generic Circuit Breaker Issues** Track 2 – Operating Reactors **Question 1:** What is the NRC and industry doing about counterfeit circuit breakers and sub-components? Recently Square D had a big recall due to this issue with molded

		case circuit breakers. **Answer 1:** The NRC staff recently took action to raise awareness of the potential for suspect/counterfeit parts and the potential for such parts to enter the nuclear industry supply chains by issuing Information Notice (IN) 2008-04, "Counterfeit Parts Supplied to Nuclear Power Plants." The Square D molded case circuit breaker issue was specifically discussed. Prior to issuing the IN, the NRC staff made a presentation, "NRC Perspectives on Dedication and Counterfeit Products," to industry representatives involved in procuring components for nuclear power plants at a meeting of the Electric Power Research Institute. The presentation discussed the Square D molded case circuit breakers. Additionally, the staff discussed some past issues and recommended reviewing some past generic communications that had documented cases of suspect/counterfeit parts. The presentation is publically available in ADAMS (accession number ML080950487). The staff also discussed the issue of counterfeit parts with the Nuclear Procurement Issues Committee. **Session Chair:** Patrick Hiland, NRC/NRR **Session POC:** Kerby Scales, NRC/NRR, tel: (301) 415-1369 e-mail: KVS1@nrc.gov
2:00 pm – 3:30 pm Salon D		**International Activities on New Reactors** Track 4 – New Reactors **Question 1:** Considering that it is typical for vendors to interact with and support regulators reviewing their designs, what processes are being implemented to facilitate vendor support for the MDEP design working groups? **Answer 1:** Membership in MDEP is limited to national regulatory authorities. The industry does have parallel efforts such as the World Nuclear Association program on international standardization, and the cooperative discussions among Codes and Standards organizations. MDEP will communicate and cooperate with such organizations, in an open manner, where appropriate. Vendors presenting their reactor designs to multiple regulators can best support the MDEP by ensuring that they communicate openly and consistently with all regulators. **Question 2:** Understanding that there is a need to firmly establish MDEP's processes for reactors, what is the likelihood of eventually extending MDEP to include fuel cycle facilities (e.g., recycling plants)? **Answer 2:** Neither the NRC nor the MDEP currently have plans to extend the MDEP program to fuel cycle facilities. **Question 3:** Is training and licensing of operators consistent from country to country; design to design? **Answer 3:** The MDEP is design-centered program, and was not intended to address operator licensing and training. As such, no data has been gathered under the MDEP program on operator licensing or training. **Session Chair:** Gary Holahan, NRC/NRO **Session POC:** Robert Elliott, NRC/NRO, tel: (301) 415-1397 e-mail: RBE@nrc.gov
2:00 pm – 3:30 pm Lower Level Entrance Marinelli & Executive Blvd		**Incident Response Experience** Track 5 – Nuclear Security, Emergency Preparedness, Fuel Cycle All questions answered onsite.

	Session Chair: Brian McDermott, NRC/NSIR **Session POC:** Janelle Jessie, NRC/NSIR, tel: (301) 415-6775 e-mail: JRB6@nrc.gov
2:00 pm – 3:30 pm Salon E	**Digital Instrumentation and Control Licensing for Power Reactors** Track 2 – Operating Reactors **Question 1:** Foreign reactors (Japan, Germany, etc...) have used digital systems for many years. Are you using this experience in lieu of re-invention which could cause unintended consequences? **Question 2:** How do back-fitting operating reactors with digital technology improve nuclear safety over existing technology? Identify specific application technology that has been identified to be problematic. What is being done to resolve the problems? **Question 3:** How vulnerable is digital I&C to an EMF burst when compared to existing technology? To what extent has this issue been reviewed? **Question 4:** Is a "mix" of old and new technology being considered? **Question 5:** To what "extreme" test standards are digital I&C actually tested? Is this testing at the component or system level? **Question 6:** A 30 minute response time is assumed for operator action. What is the basis for this 30 minute time? A reactor can be tripped at an earlier time (5 to 10 minutes). **Question 7:** To what degree have NRC / NPP and FERC cyber-security requirements and guidance been coordinated / "made consistent"? What are the plans for doing so? **Question 8:** Don't separation issues limit the use of digital equipment in non-safety related systems? **Session Chair:** William Kemper, NRC/NRR **Session POC:** Kerby Scales, NRC/NRR, tel: (301) 415-1369 e-mail: KVS1@nrc.gov
2:00 pm – 3:30 pm Brookside	**New Reactor Siting Safety Reviews** Track 4 – New Reactors Hydrology **Question 1:** What is the average area extent of hydrological exploration for siting of a Nuke? **Answer 1:** Site description consists of both surface water and ground water characterization, each one at a different scale. The surface water characterization parameters are not only estimated for the local site but consider (and depend on) the watershed in which the plant is located. Typical watersheds range from a few square miles to over one million square miles. The safety-review groundwater characterization is generally based on groundwater wells within or near the site boundary. This characterization is done for the purpose of defining groundwater pathways in the vicinity of the reactor. This information is also

		case circuit breakers. **Answer 1:** The NRC staff recently took action to raise awareness of the potential for suspect/counterfeit parts and the potential for such parts to enter the nuclear industry supply chains by issuing Information Notice (IN) 2008-04, "Counterfeit Parts Supplied to Nuclear Power Plants." The Square D molded case circuit breaker issue was specifically discussed. Prior to issuing the IN, the NRC staff made a presentation, "NRC Perspectives on Dedication and Counterfeit Products," to industry representatives involved in procuring components for nuclear power plants at a meeting of the Electric Power Research Institute. The presentation discussed the Square D molded case circuit breakers. Additionally, the staff discussed some past issues and recommended reviewing some past generic communications that had documented cases of suspect/counterfeit parts. The presentation is publically available in ADAMS (accession number ML080950487). The staff also discussed the issue of counterfeit parts with the Nuclear Procurement Issues Committee. **Session Chair:** Patrick Hiland, NRC/NRR **Session POC:** Kerby Scales, NRC/NRR, tel: (301) 415-1369 e-mail: KVS1@nrc.gov
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by the NRC for a period not to exceed 40 years from the date on which the Commission makes a finding that the acceptance criteria are met (10 CFR 52.104). A licensee may apply for renewal of that license, subject to final Commission approval, for a period not to exceed 20 years. Thus, the staff believes using a 100-year return period to determine meteorological characteristics is sufficient.

As part of the Fourth Assessment Report, the IPCC states that it is likely (>66%) that we will see increases in hurricane intensity during the 21st century. Nonetheless, the scientific debate on the impact of global warming on hurricane frequency and intensity is still ongoing and often very contentious. The NRC staff continues to monitor this debate. Should a clearer and greater scientific consensus emerge, the NRC staff would ensure that its guidance remains conservative to ensure public health and safety.

To determine hurricane wind speed site characteristics, the NRC staff considers both the historical hurricane record dating back to 1851, and engineering design standards, such as the American Society of Civil Engineers (ASCE) / Structural Engineering Institute (SEI) 7-05 Standard, "Minimum Design Loads for Buildings and Other Structures," which takes into consideration hurricane wind frequencies along coastal areas. The NRC staff is committed to ensuring that new nuclear plants sited along the coast are adequately designed to protect public health and safety.

NRC staff review guidance (NUREG-0800, Section 2.3.1) states that new nuclear power plants should be designed for climatic extremes so that they could be located at a reasonable number of sites throughout the continental United States. Should the climatic characteristics of a specific site change over time, the majority of new reactors will still have been designed to accommodate more severe weather extremes, resulting in additional safety margins.

<u>General</u>

Question 5: Is the NRC going to apply probabilistic approaches in the design basis analysis for the natural hazards other than earthquakes such as floods, tornadoes, cyclones, tsunamis to new reactor site? Please explain the analysis methods of each natural hazards.

Answer 5: General Design Criterion (GDC) 2, "Design Bases for Protection Against Natural Phenomena," in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities" states, in part, that structures, systems, and components (SSCs) important to safety be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these SSCs shall reflect, in part, appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated. Likewise, 10 CFR 52.79, "Contents of [COL] applications; technical information in final safety analysis report," states that the COL applications (FSARs) shall include the seismic, meteorological, hydrologic, and geologic characteristics of the proposed site with appropriate consideration of the most severe of the natural phenomena that have been historically report for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and time in which the historical data have been accumulated.

The analysis methods for evaluating regional climatological hazards such as tornadoes and high winds associated with cyclones are described in Section 2.3.1, "Regional Climatology," of NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP). SRP 2.3.1 states that Regulatory Guide 1.76, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants," provides guidance for selecting the design-basis tornado and design-basis tornado-generated missiles that a nuclear power plant should be designed to withstand. Note that the Regulatory Guide 1.76 design-basis tornado wind speeds for new reactors correspond to an exceedance frequency of 10-7 per year. SRP 2.3.1 also states that American Society of Civil Engineers (ASCE) Standard No. 7-05, "Minimum Design Loads for Buildings and Other Structures," can be a source for a 100-year return period 3-second gust wind speed which is also an input to the plant's wind load design. ASCE 7-05 takes into consideration the frequency of hurricane winds impacting areas along the Gulf and Atlantic coasts. However, as discussed during the RIC, to ensure the selection of site characteristics is conservative enough to account for any potential changes to the site climatology resulting from human or natural causes site characteristics (such as wind speed) should be based on the higher of either: (1) the most severe value that has been historically reported for the site and surrounding area, or (2) the 100-year return period value (e.g., from ASCE 7-05).

The limited availability of historical data creates unique challenges in formulating a sound technical basis for establishing probabilistic approaches for floods and tsunamis. Accordingly, current methods for analyzing at both floods and tsunamis are based on using a probable maximum approach, which is designed to provide a deterministically conservative result. However, the NRC in cooperation with other federal agencies is currently examining ways to develop probabilistic approaches for analyzing floods. Further, we are in the initial stages of developing internal research to look at ways of considering probabilities in assessing potential tsunami hazards.

Question 6: What is the likelihood that a review would lose its place in line if major sections of the filing lack what the NRC concludes is sufficient detail?

Answer 6: The staff expects that the applicant will provide sufficient information in order for the staff to accept the application. If an application is deficient, so that the staff cannot accept the application, then the applicant would need to revise/supplement its application and resubmit. The staff would then propose a new plan for when it would be able to conduct the acceptance review based on priorities and existing workload.

As discussed in NRC Regulatory Information Summary 2008-01, "Process for Scheduling Acceptance Reviews Based on Notification of Applicant Submission Dates for Early Site Permits, Combined licenses, and Design Certifications and Process for Determining Budget Needs for Fiscal Year 2010," the staff will allocate resources to accomplish a review based on the applicant's declaration of expected submission date. However, if during the application review, problems with responses to requests for additional information, design changes, or delays in supplemental information occur, the schedule could be significantly impacted if the staff has to perform re-work or re-plan its resources, awaiting information from the applicant. These delays may be more than day-by-day for delayed information, as the resources may also be committed to other projects such that they are unavailable when the delayed information arrives.

Seismic/Geology/Geotech

Question 7: Technical Standards for the evaluation and review of the effect of, and design basis for the surface faulting caused by a capable tectonic source near the site?

Answer 7: Requirements for the evaluation of the potential for surface faulting are provided in Appendix S to 10 CFR 50, which states "the potential for surface deformation must be taken into account in the design of the nuclear power plant by providing reasonable assurance that in the event of deformation, certain structures, systems, and components will remain functional". Further guidance for the review of the potential for surface faulting is presented in Regulatory Guide 1.208, "A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion". Appendix C of Regulatory Guide 1.208 describes the necessary investigations for assessing surface fault displacement. In addition, Appendix C (C.2.1.3) states that detailed investigations should be carried out within the site area with a high resolution in order to delineate the local geology and the potential for deformation at or near the ground surface.

Question 8: I understand that the seismic issues on high-frequency ground motion were resolved for the ESP of NPPs in the US, with applying revised USNRC SRP and ISG. For which NPPs were the seismic issues resolved? And how can I get the relevant documents on the resolution such as SERs and applicant's reports?

Answer 8: During the Early Site Permit the site specific ground motion response spectra is established. During the Combined Operating Licensing (COL) phase the site specific ground response spectra is compared to the seismic design response spectra appropriate for the NPP design. During the COL phase, the high frequency ground motion issue, if any, is resolved. Currently the NRC staff is in the process of reviewing high frequency issues associated with various design centers.

Question 9: According to SRP 3.7.2 the effects of incoherent ground motion can be taken into account in the seismic response analysis for reducing the potential effects of high frequency ground motion input. Is it applicable only to high frequency ground motion input or is it possible to broad-banded frequency ground input also? If it is not applicable to the broad-banded input, what's the reason?

Answer 9: Although the incoherency phenomena is applicable to the entire frequency range, the incoherency effects are more pronounced for the high-frequency ground motion input. The incoherency functions accepted for use in evaluations in the interim staff guide, effectively restricts reductions in the ground motion below certain cut-off frequency.

Question 10: Until when is the ISG on high-frequency ground motion effective? Is the ISG going to be incorporated in other NRC's regulatory document such as the SRP or RG sometime?

Answer 10: There is no expiration period for the ISG. After resolution of public comments, if any, the ISG will be issued in final form and its contents will be subsequently incorporated into appropriate SRP and regulatory guide sections.

Question 11: How will new information gathered or interpreted during NGA-East be factored into ongoing applications?

Answer 11: The Next Generation Attenuation Models for the Central and Eastern US (NGA-East) project was established as a five-year project. Therefore, the final product on the next generation ground motion attenuation relationships for the Central and Eastern US (CEUS) will not be expected to be available in time to affect current COL applications.

Question 12: Does the recurring nature described in the definition of "capable tectonic sources" of the Appendix A to 10 CFR 100 simply mean two fault movements within the last 500,000 years? In other words, if a fault moved two times or more only in a short period (e.g. 10,000 or 1,000 year-time period) during the last 500,000 years, but with no other movements during the rest of the period, do you still call the fault a capable tectonic source?

Answer 12: Appendix A of 10 CFR 100 currently applies only to nuclear power plants sited prior to January 10, 1997. New reactor applications refer to 10 CFR 100.23 and Regulatory Guide 1.208, titled "A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion", with regard to requirements and guidance for characterizing potential capable faults. In accordance with the definition of a capable fault in RG 1.208, a fault that has moved twice during the past 500,000 years would be considered capable.

		Session Chair: Nilesh Chokshi, NRC/NRO **Session POC:** Zahira Cruz-Perez, NRC/NRO, tel: (301) 415-3808 e-mail: ZLC@nrc.gov
2:00 pm – 3:30 pm Salon F-H		**Operating Reactor Licensing** Track 2 – Operating Reactors **Question 1:** Even after successful acceptance reviews and months of technical review, some LARs still seem to be "hung up" by continuing Requests for Additional Information within a few weeks of the licensee "need date". In some cases the RAIs question the current plant configuration not the proposed configuration. What can be done to minimize these situations? **Question 2:** When does the clock start for your timeliness metric – before or after the acceptance review is completed? **Question 3:** How will the use of precedence (previously acceptable submittals) be impacted by the acceptance review process? **Question 4:** Are efforts being taken to minimize changes in submittal reviewers? **Question 5:** What are your thoughts on pre-submittal meetings on complex LARs or LARs that have no clear precedence? **Question 6:** When will the NRC's internal procedure for acceptance reviews be available to the public as a final document? **Question 7:** Will the LIC on acceptance reviews be a "living document" at least for an initial roll-out period? I expect industry (LATF) will give you feedback that would help improve the process. **Question 8:** Have you established timeliness metrics for completing the new acceptance review process? If so what are they? **Question 9:** What tools will reviewers be given to help ensure standardized acceptance reviews? What degree of management oversight is planned to confirm that different reviewers come to similar conclusions? **Session Chair:** Catherine Haney, NRC/NRR **Session POC:** Peter Bamford, NRC/NRR, tel: (301) 415-2833 e-mail: PJB1@nrc.gov
3:30 pm – 4:00 pm		Break
4:00 pm – 5:30 pm Brookside		**LOCA: Cladding Embrittlement** Track 3 – Reactor Research **Question 1:** Confirm that ANL [Argonne National Laboratory] uses the Cathcart-Pawel oxidation relation and JAEA [Japan Atomic Energy Agency] uses the Baker-Just relation to calculate ECR [Equivalent Cladding Reacted]. **Answer 1 [ANL]:** That is correct. ANL uses the Cathcart-Pawel oxidation relation and JAEA uses the Baker-Just relation to calculate ECR. **Question 2:** Other than being nice to have more data, what specific deficiencies do you see in NRC's rulemaking strategy that require more data prior to changing the 17% limit to a performance-based limit that will apply to all zirconium alloys, including M5?

Answer 2 [Electric Power Research Institute (EPRI)]: That depends to a great extent on what NRC will propose for the revised criteria and how it will propose addressing issues such as those raised by IRSN (at this meeting) or by the ongoing experimental program at Halden. Without a clearer understanding of the potential impact of phenomena such as the uptake of oxygen from the ID surface of the cladding, the brittle behavior of the ballooned region, extent of fuel relocation into the ballooned region, dispersal through the burst opening, etc. the regulatory tendency is likely to be to address these phenomena in an unnecessarily conservative manner, assuming that they are all 100% relevant.

Question 3: Initially, the USAEC [U.S. Atomic Energy Commission] chose ductility rather than strength because LOCA [loss-of-coolant accident] loads were essentially unknowable. What is the basis for the maximum load assumed in Japan?

Answer 3 [JAEA]: The maximum load has been investigated experimentally in JAEA and Japanese industries. The maximum load of 540 N (50 kgf), used in the JAEA's thermal shock test, is the highest value measured in these experiments. Most data are ~200 N or lower, but we adopt the 540 N as a conservative, bounding condition.

Question 4: The [Halden] 650.4 rod was about 90 GWd/MtU. Would IRSN support the testing of a more reasonable burnup (60 – 70 GWd/MtU) to remove excessive pellet degradation from the considerations?

Answer 4 [Institut de Radioprotection et de Sûreté Nucléaire (IRSN)]: The Halden test matrix should of course be complemented by tests performed at burnups around 60-70 GWd/MtU. Testing at higher burnups will nevertheless remain useful for our understanding of burnup effect on relocation and dispersion, and should not be excluded from our investigation field.

Results from other experimental programmes (ANL, FLASH…) have shown a high fragmentation of fuel in the burst region for burnup around 50 GWd/MtU, thus HALDEN IFA 650-4 results shall not be disregarded.

Question 5: Experimental data show hydrogen in [the] burst region to be the main problem/uncertainty. In 50.46 proposed rulemaking, has NRR considered (a) limiting PCT < T_{burst} and (b) reducing transition break size to ~SBLOCA? BE methods should show PCT < T_{burst} and the need to understand H_2 effects would go away.

Answer 5 [NRC]: NRC has no plans to change the peak cladding temperature (PCT) limit of 1204°C (2200°F) in 10 CFR Part 50.46(b). Since cladding rupture occurs at substantially lower temperatures (e.g., 800°C - 900°C), the change proposed in this question would be very limiting. For current operating reactors, it is doubtful that even best-estimate methods could ensure that PCT < T_{burst} for the large break loss-of-coolant accident. For smaller break sizes (i.e., below transition break size), such behavior is possible. However, NRC has no plans to reduce transition break size to account for cladding behavior.

Question 6: We all understand that the mechanism of embrittlement is related to the beta layer. Do you see any problem with NRC's using an empirical correlation that is related to calculated oxidation?

Answer 6 [IRSN]: Defining a safety criterion based on a calculated oxidation rate is consistent with the original approach of the 1973 criterion, but will require (a priori) a calculation methodology that is conservative with regard to all the transients and physical mechanisms that could affect the clad ductility.

Since embrittlement is directly related to the beta layer and its content of O and H but is not directly related to the cladding oxidation rate, the approach of USNRC presents two types of risk:

- the first is to forget or fail to consider some possible combinations of physical phenomena when fixing the maximum acceptable oxidation rate and thus create a non-conservative criterion,
- the second is that a too simple criterion formulation might lead, in some situations, to an excessive conservatism.

Question 7: You suggested using impact tests and using evaluation of strength as a criterion. This was specifically turned down in the "Statement of Considerations" in the 1973 Rulemaking as very impractical. How would you deal with thermal stress during quench and vibrations?

Answer 7 [EPRI]: We suggested that strength as measured by impact resistance or by survivability of quench under an axial load (such as done in Japanese experiments) could be used as an alternative to ductility. In addition to giving us some margin, a strength-based approach is likely to give us a means for addressing the brittle ballooned region. The thermal stresses that occur during quench are specifically addressed by the quench survivability tests.

Question 8: With respect to the issue of "chips and fines" released during RIA tests, what is the NRC's position on fuel dispersal from a burst region during a LOCA test conducted by [Halden and cited by] IRSN?

Answer 8 [NRC]: The concept of "chips and fines" was used in the Research Information Letter 0401 as a possible explanation for data scatter in the onset of cladding plastic strain measurements in RIA tests. It does not refer to fuel relocation and dispersal observed in the Halden LOCA tests cited by IRSN. NRC is reviewing the relocation and dispersal issue in the context of 10 CFR 50 Appendix K (LOCA models) rather than 10 CFR 50.46(b) (cladding embrittlement).

Question 9: With respect to CABRI RIA testing, what is the hydrogen concentration in the various tests? Were SEM analyses done for the various test samples? Has this information be[en] generally released to the industry?

Answer 9 [IRSN]: The cladding hydrogen concentration of fuel rods depends mainly on the clad and of the burnup. A large spectrum of local and average hydrogen concentrations has been covered by fuel rods tested in CABRI: up to 1370ppm and locally up to 2000ppm. Hydrogen concentrations have been systematically measured and results have been made available for all the CABRI programme partners.

Question 10: What was the maximum fuel burnup tested?

Answers for 10:

[ANL] Rod average burnups are listed on Slide 13 in the ANL presentation. With regard to high-burnup cladding alloys tested to date, the highest are: about 70 GWd/MTU rod averaged and about 76 GWD/MTU local burnup.

[EPRI] The Halden test (IFA 650.4) that resulted in fuel dispersal used a fuel specimen with a butnup of 91.5 GWd/MTU.

[IRSN] Fine fuel fragmentation has been observed in the HALDEN test 650-4 for 91.5 GWd/MtU, but also in tests FLASH-5, ANL ICL-2 and Halden test 650-5 respectively for 50, 56 and 83 GWd/MtU.

[JAEA] ~76 GWd/t at JAEA.

Question 11: Is there a performance-based hypothetical burnup limit for Zr-2 and Zr-4 fuels?

		Answer 11 [NRC]: No. As long as hydrogen absorption is controlled to a low level, no burnup limit is foreseen for any cladding alloy. On the other hand, the recent Halden test (IFA-650.4) suggests that there might be a burnup limit associated with the fuel as opposed to the cladding. In that test, conducted at a burnup of about 92 GWd/t, a very large "rim" structure developed in the pellet. This resulted in a substantial volume of particulate fuel, which flowed out of the burst opening under gravity. Only small "rims" are seen at the current operating limit of 62 GWd/t, so there is probably a need for a technically-based limit at some burnup around or above the current operating limit. **Session Chair:** Farouk Eltawila, NRC/RES **Session POC:** John Voglewede, NRC/RES, tel: (301) 415-7415 e-mail: JCV@nrc.gov
4:00 pm – 5:30 pm Salon F-H		**Fire Protection: Recent Achievements and Remaining Challenges** Track 2 – Operating Reactors All questions answered onsite. **Session Chair:** Mark Cunningham, NRC/NRR **Session POC:** Chuck Moulton, NRC/NRR, tel: (301) 415-2751 e-mail: CEM4@nrc.gov
4:00 pm – 5:30 pm Salon A-C		**Nuclear Security** Track 5 – Nuclear Security, Emergency Preparedness, Fuel Cycle **Question 1:** In terms of radiological sabotage including intended aircraft crash, how does the NRC deal with security, safety, and operational interface? **Question 2:** Will the Part 73 rulemaking NRC is pursuing affect the RRTT concept? **Question 3:** Does incorporation of answers to Security Frequently Asked Questions (SFAQ) in security plans automatically fall under 50.54p? **Question 4:** Regarding the Palisades Nuclear plant, has there been/is there a "No Go" exclusion zone in Lake Michigan? If not, why not? If yes, why are recreational watercraft allowed to anchor immediately offshore from Palisades? How is the presence of a "No Go" zone communicated to the general public, given a lack of signage in such a highly trafficked are of Lake Michigan? **Question 5:** Where do you see the link between the security plan and an emergency plan in a COLA submittal? **Question 6:** Has NRC improved security guard/security director hiring screening safeguards in light of the hiring of William "Zeke" Clark at Palisades by Consumers Energy and his retention by Entergy as reported by Esquire magazine in May 2007? If yes, what improvements have been implemented? If not, why not? **Session Chair:** Dan Dorman, NRC/NSIR and Trish Holahan, NRC/NSIR **Session POC:** R. John Vanden Berghe, NRC/NSIR, tel: (301) 415-7142 e-mail: RJV@nrc.gov
4:00 pm – 5:30 pm Salon D		**New Reactor Technical Issues/Systems** Track 4 – New Reactors All questions answered onsite. **Session Chair:** Frank Akstulewicz, NRC/NRO **Session POC:** Hanry Wagage, NRC/NRO, tel: (301) 415-1840 e-mail: HAW2@nrc.gov

4:00 pm – 5:30 pm Lower Level Entrance Marinelli & Executive Blvd		**Incident Response Experience** Track 5 – Nuclear Security, Emergency Preparedness, Fuel Cycle All questions answered onsite. **Session Chair:** Brian McDermott, NRC/NSIR **Session POC:** Janelle Jessie, NRC/NSIR, tel: (301) 415-6775 e-mail: JRB6@nrc.gov
4:00 pm – 5:30 pm Salon E		**Region I, II, III and IV Breakout Session** Track 6 – Regional Breakout All questions answered onsite. **Session Chair:** Bruce Mallett, NRC/EDO **Session POC:** Randy Musser, NRC/R-II, tel: (404) 562-4603 e-mail: RXM1@nrc.gov

Thursday, March 13, 2008

8:00 am – 9:30 am Salon A-C	**Spent Fuel Storage and Disposal** Track 5 – Nuclear Security, Emergency Preparedness, Fuel Cycle All questions answered onsite. **Session Chair:** William Brach, NRC/NMSS and Lawrence E. Kokajko, NRC/NMSS **Session POC:** Yen-Ju Chen, NRC/NSIR, tel: (301) 492-3238 e-mail: YJC@nrc.gov
8:00 am – 9:30 am Salon F-H	**Thermal-Hydraulic Code Development and Applications** Track 3 – Reactor Research All questions answered onsite. **Session Chair:** Stephen M. Bajorek, NRC/RES **Session POC:** Daniel Forsyth, NRC/RES, tel: (301) 415-5674 e-mail: DCF1@nrc.gov
8:00 am – 9:30 am Salon D	**New Reactor Licensing: Matching Expectations and Reality** Track 4 – New Reactors **Question 1:** When would it be appropriate to use Part 50 in lieu of Part 52 for licensing a commercial nuclear power plant? **Answer 1:** The applicant has the option to decide which licensing process is more appropriate in applying for a license. Part 50 is a two-step licensing process in which a Construction Permit can be issued based on a review of preliminary design information, site suitability, and environmental impacts. As construction is nearing completion, the applicant then applies for an Operating License, which includes final design information, conditions for plant operation, and supporting programs. Under the Part 50 process, a hearing on the Construction Permit is mandatory. There is a hearing on the Operating License application only if a petitioner for intervention demonstrates standing and proffers at least one admissible contention. Additionally, because construction is started before the design of the plant is completed, historically schedule slippages and cost increases have been the norm where there was the need for design changes and rework. The NRC established an alternative one-step licensing process in 10 CFR Part 52, “Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants,” which the staff believes is a more predictable and efficient licensing process. This process combines the Construction Permit and the Operating License (COL) with conditions for plant operation. However, the Part 52 process requires an essentially complete design as a condition for applying under Part 52. The Part 52 process has the advantages of one hearing on plant design, site suitability, and environmental impact. There is an opportunity for an additional hearing following

		because a significant part of the licensing review has already been conducted and fewer issues need to be reviewed as part of the COL. **Session Chair:** Thomas Bergman, NRC/NRO **Session POC:** Meena Khanna, NRC/NRO, tel: (301) 415-2150 e-mail: MKK@nrc.gov and James Steckel, NRC/NRO, tel: (301) 415-1026 e-mail: JAS13@nrc.gov
8:00 am – 9:30 am Lower Level Entrance Marinelli & Executive Blvd		**Incident Response Experience** Track 5 – Nuclear Security, Emergency Preparedness, Fuel Cycle All questions answered onsite. **Session Chair:** Brian McDermott, NRC/NSIR **Session POC:** Janelle Jessie, NRC/NSIR, tel: (301) 415-6775 e-mail: JRB6@nrc.gov
8:00 am – 9:30 am Salon E		**Risk Informed Regulatory Activities** Track 2 – Operating Reactors **Question 1:** With the change in Commissioners, is there continued support for risk based initiatives? **Question 2:** Are there disagreements on results of RI decisions between NRC and utilities and how do you resolve them? **Question 3:** The new reactors (AP1000, EPR, ESBWR, etc.) may have very low core damage frequencies. Will the NRC revise the risk criteria? **Question 4:** **Question 4:** What will be the role of PRA in addressing issues associated with "life beyond 60," and aging issues? **Question 5:** How could we use risk-informed considerations to account for uncertainty in instrument set point-related technical specifications? **Question 6:** If you change surveillance intervals that will change the component of instrument uncertainty. Uncertainty will increase, therefore set points in tech specs need to be more conservative. Please comment. **Question 7:** Please provide some examples where a "business-case" PRA provides important and different results than a "safety-case" PRA. **Question 8:** The presentations focused on PRA modeling standards, etc. Where do you

		Session POC: Andrew Howe, NRC/NRR, tel: (301) 415-3078 e-mail: AJH1@nrc.gov
8:00 am – 9:30 am Brookside		**Research Findings Related to NRC Revised Source Term (NUREG-1465)** Track 3 – Reactor Research **Question 1:** Clad oxidation under air ingress was observed to be peculiar. Is this different from the behavior observed in air oxidation experiments performed under the EU framework program a few years ago and in the ANL program performed subsequently? **Answer 1:** Clad oxidation under air ingress above 1000 °C is indeed more violent than under steam, for two reasons: the reaction between zirconium and oxygen is more energetic and nitrogen appears to make the oxide layer formed during the oxidation process unstable (break-away). In addition, zirconium nitrides may violently react with oxygen and burn. The data that we obtained from the International Source Term Program are confirming ANL results and are giving some insights on the break-away phenomenon. **Question 2:** Have the Phébus tests been compared to, or correlated with, the INL LOFT tests of the early 80's? Does Phébus simulate the typical proportionality of commercial PWR systems, structures, & components; and does Phébus use commercial PWR materials? **Answer 2:** There are not many differences between LOFT and Phébus data as far as fuel degradation and volatile fission product release rate are concerned. Phébus went further in fuel degradation than any experiments performed before and is giving quite new results concerning fission product transport and behaviour in containment. In particular, the data on iodine volatility in the containment are unique. Note that there was no containment model in the LOFT tests. Phébus simulates the fuel, cladding and fission product inventory closely. The control rod material inventory is slightly larger but this is not felt as a problem as it is already largely in excess in the reactor case as compared to the fission product inventory. The length and the surface of the circuit are certainly not at scale, but fission product deposition along the circuit is not that large. The volume and the cooled surfaces (inner condensers located at the centre of the model) of the containment model are at scale; the outer surface of the containment is heated in order to avoid any steam condensation and be as neutral as possible regarding fission product deposition. The sump volume and surfaces are not totally at scale, the option taken being to favour representative radiation level conditions. The fuel rods used in Phébus FPT-1, FPT-2 and FPT-3 tests came from the pressurized water reactor BR3, in Belgium; they were typical UO2 Zircaloy 4 cladded fuel rods with a little bit more than 20 GWd/TU. The control rod simulator was made of typical SIC alloy used in PWRs, stainless steel cladded and located in a Zicaloy 4 guiding tube. **Session Chair:** Richard Lee, NRC/RES **Session POC:** Michael Salay, NRC/RES, tel: (301) 415-5603 e-mail: MAS10@nrc.gov
8:00 am – 9:30 am White Flint Amphitheater		**Increased Openness and Transparency in NRC Security Inspection Programs** Track 5 – Nuclear Security, Emergency Preparedness, Fuel Cycle All questions answered onsite. **Session Chair:** Rich Correia, NRC/NSIR **Session POC:** Paul W. Harris, NRC/NSIR, tel: (301) 415-1169 e-mail: PWH1@nrc.gov

9:30 am – 10:00 am		Break
10:00 am-11:30 am Salon A-C		**Ground-Water Contamination Assessments for NRC-Licensed Facilities** Track 3 – Reactor Research

Question 1: Given that ground-water contamination isn't truly a regulatory issue, what more action will or should the NRC staff take on this issue?

Answered by panel member during the session.

Question 2: NRC allows 1 million picocuries of tritium per liter in ground-water, while EPA's Safe Drinking Water Act limit is 20,000 picocuries per liter. The State of Colorado and California however have standards in place 40 to 50 times more protective than the EPA's. Given this, given NAS BEIR VII's recognition that even low level radioactive exposure carries health risks, and given chronic public exposure to tritium and the enhanced risk of organically bound tritium, how can NRC continue to downplay tritium's health risks?

Answered by panel member during the session.

Question 3: The foundation design of the NPP should take into account local ground-water flow and transport. Why then, should there be uncertainty in ground-water flow and transport determination for the local hydrologist?

Answer 3: Ground-water flow and transport models are abstracted representations of complex ground-water flow systems. These models are developed from discrete samples of information collected at specific locations and times which are points within a distribution of possible values. Ground-water flow and transport processes by their nature are dynamic and vary with time as a function of episodic events such as recharge and drainage. Because of the paucity of site data on these processes, as well as geologic and anthropogenic features and events affecting ground-water flow, there is always a possibility that there are alternative interpretations of this limited information (i.e., alternative conceptual models) or that sampled values are extremes of the distributions. These inherent uncertainties in the data and the abstracted models will exist for any hydrologist, even one familiar with the site and plant design.

Question 4: In relation to the EPRI graded approach, how do you determine "high

Question 5: Is there an ASTM, EPRI, or ANSI standard established for monitoring well design, installation, and operation depending on hazard (planning for radionuclide vs. organic hydrocarbons detection)?

Answer 5: I am not aware of a published standard for monitoring wells, based upon hazard (there are ASTM standards on design, installation and operation of ground-water monitoring wells). Many organic chemicals are immiscible in water; some have densities greater or less than water and, therefore, either sink or float in the water column if sufficient volume is spilled or leaked such that a non-aqueous phase is present. Radioactive liquids are miscible solutions (or tritiated water) that does not form a non-aqueous phase in water. The density of these liquids generally does not need to be considered. It is the responsibility of the hydrogeologist to design monitoring wells based upon the physical and chemical properties of the contaminants of interest and the physical properties of the strata through which the contaminants are transported.

Question 6: According to the Lessons Learned Task Force's 26 consolidated recommendations, NRC should develop or revise or provide guidance to the industry. Are there any newly developed or revised or provided NRC regulations/guidance to address remediation?

Answer 6: As part of a proposed rulemaking known as "Decommissioning Planning" (RIN 3150-AH45), NRC developed guidance on identifying unplanned radioactive releases to the environment. The guidance suggests representative criteria by which licensees can determine if they should conduct prompt remediation of such an event or increase the Decommissioning Fund to cover the cost of remediating later. There is no specific guidance on how licensees should proceed with remediation in either case.

Question 7: Are there any reported instances of ground-water contamination due to leakage from nuclear power plant structures and systems through reactor foundation concrete basemats?

Answer 7 (and 11): In 2006, the Nuclear Energy Institute (NEI), in conjunction with the nuclear industry, developed a questionnaire to be completed by operators of nuclear power plants. The purpose of this questionnaire was to establish the baseline for monitoring and protecting the groundwater at power plants. In the questionnaires, a number of nuclear power plants identified leaks that had the potential to contaminate ground water. The power plants also indentified any remedial actions taken. The completed questionnaires from all the nuclear power plants can be found at: http://www.nrc.gov/reactors/operating/ops-experience/tritium/plant-info.html.

In 2007, NEI submitted the Industry Ground Water Protection Initiative – Final Guidance Document, NEI 07-07. While the Ground Water Protection Initiative (GPI) is a voluntary program, upper management at the various nuclear utility companies have endorsed the program. As stated in the Executive Summary: "It is expected that this Initiative will be implemented by each member company currently operating or decommissioning a nuclear power plant and by each member company constructing a new plant after year 2006. In the event that new or amended NRC regulations are enacted that address ground water protection or inadvertent releases of radioactive liquids, this Initiative should be revisited by the Nuclear Strategic Issues Advisory Committee."

The GPI contains a number of objectives which include 1) evaluate systems, structures or components and work practices for which there is a credible mechanism for radioactive material to reach the ground water; 2) establish an on site ground water monitoring program which can ensure timely detection of any inadvertent radioactive material releases to the ground water; and 3) establish a remediation protocol which will prevent migration of licensed material off site and will minimize decommissioning impacts.

Licensees have committed to having implemented the GPI and a self-assessment of compliance completed by the end of August 2008. NRC Regional Inspectors will begin verifying implementation of the GPI by inspection starting in September 2008.

Question 8: Detection of ground-water contamination in wells suggests that failure has already occurred. Have you thought of detection at the interface between the source and soil BEFORE contamination becomes a problem?

Answer 8: Yes. Demonstration of compliance with 10 CFR 20.1406 (a) and (b) is based on consideration of design features and operating procedures at the time of license application that will minimize contamination. The guidance in Regulatory Guide 4.21 suggests that the approach for compliance should incorporate design features to prevent leaks, provide systems to detect leaks as soon as possible if prevention fails, and provide plans to deal with leaks once detected.

Question 9: You told us about actions on how to deal with contamination after the fact. Do you see a role for prevention? And what has EPRI done on that front?

Answer 9 (and 11): EPRI has developed a process for calculating a "Priority Index" for each structure, system, component (SSC) or work practice that contains radioactive liquids for which a plausible mechanism exists for release to the environment. The process is intended to identify those SSCs and work practices with the highest potential for a leak or spill, so that preventative measures can be concentrated on these areas. Preventative measures may include conducting preventative maintenance, increasing surveillance and inspection, conducting integrity testing, installing secondary containment, improving leak detection and overfill protection systems, installing ground water monitoring systems, altering work practices, or other measures.

Question 10: You have rightly stressed the importance of communication. Do we have an effective answer to a concern about community drinking ground-water source that has been contaminated by radioactive elements?

Answer 10: The NRC and EPA regulations establish standards for protection against radiation. For example, the NRC establishes standards for radioactivity in liquid effluent discharges from our licensed facilities, and the EPA establishes standards for public drinking water sources. The NRC maintains and enforces these requirements and radiation limits to ensure public safety. The NRC's radiation limits account for the principal exposure pathways (e.g., ingestion, inhalation, etc.) that an individual can be exposed to from effluent discharges from our regulated facilities, including an individual drinking water that may have been affected by radioactive effluent discharges. The NRC ensures these standards are not exceeded by requiring our licensee's to monitor radioactive effluent discharges and to limit discharges to within acceptable standards and by requiring our licensee's to verify the adequacy of effluent programs by monitoring radioactivity in the environment. By ensuring that the limits are not exceeded, we ensure that the public remains safe and that drinking water sources are safe to consume.

Question 11: You discussed communications and remedial action. Has any action been taken to prevent some of the more common "leaks?"

Yes, please see Answers 7 and 9

Question 12: For new reactor applications, how does NRC ensure that local hydrology is taken into account if the detailed plant design is not completed?

Answer 12: RG 4.21 will contain regulatory guidance on methods acceptable to the NRC to meet 10 CR 20.1406, including the effects of local hydrology on contamination of the site and environment. In particular, RG 4.21 discusses the development of a conceptual site model (CSM) to evaluate potential ground-water transport pathways, and the need to continue to update and evaluate the CSM following construction to evaluate the impacts to the local hydrology of construction activities. Early Site Permit reviews, as outlined in Review Standard RS-002 “Processing Applications for Early Site Permits,” also consider local hydrology.

Question 13: Given that some remediation techniques leave the contaminant at the site, what is the key uncertainty and what does this imply for post-remediation monitoring?

First Answer to 13: The key uncertainty is what site-specific processes, events, and conditions could later mobilize the contaminants during the post-remediation period. For instance, a change in ground-water chemistry due to ground water recharge caused by flooding or introduction of an organic contaminant could create an adverse condition to re-mobilize the contaminant.

Second Answer to 13: No remediation technique short of excavation can remove every atom of uranium (U) from the site. The combination of reductive precipitation with FeS encapsulation does not need to reach every atom of U because by addressing the vast majority of the impacts the remaining residual is diluted to below the regulatory limit. Also, by emplacing a stochiometric excess of reduced iron sulfide minerals which will eventually oxidize to highly sorptive ferric iron minerals, any residual U is unlikely to travel any significant distance before being sorbed. Therefore, the key uncertainty and the primary performance monitoring parameter for a remedial technology like this is the adequate distribution and requisite mass of reduced iron sulfide minerals.

Question 14: [Boyce] Clark, in your remedial action concept, do you know what can happen after 1000 years? Is the environment oxidizing or reducing in the long term?

Answer 14: The current baseline conditions at the site are slightly oxidizing. Therefore, upon cessation of organic carbon injections, conditions will return to oxidizing and the reduced iron sulfide minerals will begin to oxidize. The targeted mass of iron sulfide to be emplaced is based on the groundwater velocity and dissolved oxygen concentration. We target an iron sulfide concentration that will effectively buffer the redox of the incoming groundwater for 1000 years. As the iron sulfide minerals are oxidized they will likely passivate the surface of the grains providing a physical barrier to the reduced uraninite against oxygen. This process will dramatically extend the stability period beyond 1000 years. There will be small areas throughout the plume that uranium will dissolve. However, the mass of reduced iron sulfide minerals that undergo oxidation along with the uraninite will sorb any released uranium.

Question 15: Given that the in-situ precipitation process may add large amounts of Fe and SO4 to the water, does subsequent dissolution/oxidation of the iron sulfides affect ground-water quality?

Answer 15: The slow oxidation of iron sulfides will create ferric oxyhydroxide and oxidized sulfur forms, including thiosulfate; this eventually oxidizes to sulfate. The natural buffering capacity of the aquifer, due to the presence of calcite and the generation of additional alkalinity and carbonate minerals during biological sulfate reduction, will negate any pH change during oxidation of the iron sulfide.

Question 16: Would the in-situ treatment be applicable to the kinds of radionuclides we have seen at nuclear plant site, including Tritium, Strontium-90, and Cobalt-60?

Answer 16: In-situ precipitation is applicable to elements that can form a solid phase through redox manipulation or can be encapsulated within an iron mineral matrix. Other radionuclides that are amenable to this process include technetium-99; this radionuclide is very similar to uranium in that it is soluble in its oxidized form (TcO4-) and insoluble in its reduced form (TcO2). Recent work published in the peer-reviewed literature has shown that reduced iron minerals play a large role in keeping TcO2 immobilized. Strontium and cobalt are not redox-active; however, they can sorb from groundwater onto iron mineral phases; some of these iron minerals can be made more sorptive through in-situ biological activity.

Question 17: In evaluating the long-term performance of the remediation process, how is the change in stresses in the materials and geologic strata beneath the major structures considered? The removal of structures will change the transport characteristics.

Answer 17: To my knowledge, no attempt has been made to date to monitor change in stress within the geologic strata beneath major structures when the structures are removed. Numerical modeling may be a useful tool to predict changes in the stress field when structures are removed.

Question 18: In view of the long life of radionuclides, is monitored natural attenuation (MNA) really a viable remediation technique?

Answer 18: The efficacy of this approach is still under study. EPA is developing guidelines for MNA of radionuclides and metals which are schedule for release in 2008. The "M" in MNA is for monitored. It is critical that Performance Indicators be identified, monitored and analyzed to demonstrate that the contaminant plume is attenuating. It should be noted that mineral deposits containing long-lived radionuclides have persisted for time frames up to billions of years. One difficulty is that many sites needing remediation have residual contaminants in the shallow subsurface subject to more variable conditions than mineral deposits that are often found at depth. Thus establishing the proper conditions and then demonstrating that those conditions will persist is technically challenging.

Question 19: Are there attempts made to monitor downstream of the highly contaminated area for (1) agriculture products; (2) cattle (milk and meat); (3) live farms (shrimp, salmon, etc.)?

Answer 19: All nuclear power plants are required to have a radiological environmental monitoring program (REMP) to ensure compliance with NRC regulations in 10 CFR Parts 20 and 50. As part of the REMP, licensees will sample and analyze ground water, milk from local dairies if the milk animals are on pasture, and various types of vegetables and vegetation grown in the vicinity of the plant. Plants are required to do perform a land use census in the five-mile radius of the plants to identify such things as nearest residents, gardens, milk animals. Any significant change identified in the land use census will be evaluated and may result in changes to the REMP. Environmental reports are submitted to the NRC in the form of an annual report. Licensees must also monitor and report the amount of radionuclides released in all gaseous and liquid effluents from their facility and report the results to the NRC in an annual report. The reports for environmental samples and effluent samples for the years 2005 and 2006 can be found on the NRC public web site (http://www.nrc.gov/reactors/operating/ops-experience/tritium/plant-info.html). The 2007 data from commercial nuclear power plants effluent and environmental monitoring will be reported in May, 2008 and will be available on the NRC web shortly thereafter.

Question 20: Long term safety case assessments at Yucca Mountain show importance of colloids in more rapid mass transport issues in the vadose zone. Has this been considered in site assessments/recommendations?

Answer 20: A number of studies have been conducted to look for colloids in remediated systems and they have been rarely, if ever, found. Research on in-situ bioremediation is looking at colloidal transport at various DOE sites. NRC staff is following this work to determine if colloidal transport can affect long-term performance.

Question 21: What design changes will be used for new reactors to prevent ground-water contamination in the future – especially from pool storage?

Answer 21: RG 4.21 will contain regulatory guidance on methods acceptable to the NRC to meet 10 CR 20.1406, including system design and operating practices to minimize and detect leakage from spent fuel pools. For example, RG 4.21 discusses designing spent fuel pools with clearly defined zones that have the capability to detect and quantify small leakage rates from each zone.

Rulemaking Program: Looking Below the Surface

Track 2 – Operating Reactors

Session Chair: Sher Bahadur, NRC/RES
Session POC: Adam Schwartzman, NRC/RES tel: (301) 415-8172 e-mail: ALS2@nrc.gov

10:00 am-11:30 am
Salon E

Question 1: At a previous RIC, Skip Bowman said that if he NRC wants to require actions by the industry, the NRC should do it via rulemaking. Is this still the NEI position?

Answer 1: It is important that the NRC follow its regulatory process and not impose new requirements through informal means, and without conducting a backfitting analysis.

Question 2: What is NEI's view on when industry should be involved in the Tech Basis development? Would it be later in the rulemaking process or earlier?

Answer 2: We believe that all interested stakeholders should be invited earlier in the process; in fact, I would say at the onset of the development of the technical basis.

Question 3: Is the Regulatory Guidance from NEI-07-06 available to the public/NRC personnel?

Answer 3: NEI 07-06 describes the regulatory process and is available on the NEI member website and also on request by the public or NRC staff. I have attached a copy which can be made available by the NRC in their RIC materials.

Question 4: Please define "ROP."
Answer 4: Reactor Oversight Process. This is the NRC's method of inspecting and assessing licensees' performance.

Question 5: Please provide a recent example of a new rule which drove industry away from safety.

attached a copy which can be made available by the NRC in their RIC materials.

Question 8: Does the NRC intend to write future rules so someone, other than a lawyer, can understand what is required?

Answer 8: It is the NRC's intention that everyone understands a final rule when they read the rule. They are carefully written to leave out misunderstandings by those licensees who have to follow them. What may be difficult for the general public at times is the degree of technical information inserted in the language. Whenever possible, members of the public should read the background information in the Federal Register notice issued when the rule is published. This background information explains the need for the rule and also contains a paragraph by paragraph explanation of the requirements in the rule language.

Question 9: Will the agency ever go to Question-Answer format in its regulations? Because it is easier to understand, plain English usually is used, government initiative. If not in our regulations, why not in guidance documents?

Answer 9: The Agency is always looking for efficient ways to accomplish its mission, but a Question-Answer format is not being considered at this time. Whenever guidance documents are written, NRC does its best to involve stakeholders in the process so that they can understand the guidance clearly for rule implementation. Guidance documents contain the rule language in a wider context while attempting to clearly state the requirements.

Question 10: Executive orders versus rules – please explain impact on regulations and implementation.

Answer 10: Executive Orders arrive at the NRC from the executive branch of government. To implement these Orders, the NRC sometimes has to issue new rules through the same process we have currently. The impact of each rule carries its own weight and their implementation is via standard mechanisms used by the NRC.

Question 11: Is there agency consensus that technical basis must come first? What can be done to increase public confidence that comments will be considered seriously? Will there be public participation in attempts to resolve the rulemaking process?
Answer 11: It is the NRC's desired outcome as outlined in internal procedures to have the Technical Basis completed before entering the proposed/final rule phase. This outcome makes the whole process flow more efficiently. It is the NRC policy to be receptive to public participation, especially as stakeholders understand the technical issues clearly and are willing to share their input. Nevertheless, it is the NRC's staff responsibility to generate the Technical Basis for the rule.

Question 12: If technical basis work were completed prior to rulemaking as you suggest, could this lessen or even eliminate the need for guidance to be issued after the rule is finalized (e.g. the better the rule, the less guidance is needed to implement it)?

Answer 12: No. A well defined technical basis makes the process of issuing a rule more efficient and effective. The technical basis does not eliminate the need for implementation guidance after a rule is issued. Implementation guidance is typically contained in Regulatory Guides, which provide examples/ways that licensees can comply with a particular regulation.

Question 13: Does UCS agree that the Emergency Planning Rulemaking is an example of "the good"?

Answer 13: No. The emergency preparedness rulemaking process should never have happened as it did and must never be replicated. It was grossly unfair to the public. The NRC met with the industry representatives repeatedly to negotiate what the industry wanted. The NRC met with the public once, and it's not clear from the rulemaking package that the NRC even considered any of the input received from the public during that sole meeting. In addition, we have evidence that the NRC staff shared information with industry representatives that it did not share with the public. The process was a disgraceful sham.

Question 14: For UCS, which is it? Bad rules or bad enforcement? You started by saying the rules are fine and that NRC needed to better enforce. Then you criticized the rulemaking process as a step before legal action.

Answer 14: I concede the apparent disconnect. But the invisible bridge is the public's desire to use the courts to do what the NRC won't do - enforce the regulations. The public cares little about whether the rulemaking process sets the bar at 9 or 10, since the NRC won't enforce either. The public must turn to the courts because the NRC won't do its job.

Question 15: For UCS, you assume we never get above 10. What if the new rule can get us to 12 because existing regulations provide the NRC purpose and goal, and the rulemaking will make it better?

Answer 15: I intended that "10" be a relative value. There have been rulemakings that raised the bar, in essence going from "10" to "12." The maintenance rule (10 CFR 50.65) is such an example.

Question 16: For UCS, regarding your fire protection issues and your compensatory actions – allowed. Are you suggesting that an "administrative" limiting condition for operation should always have a shutdown requirement or a time limit even if more compensatory measures provide an adequate level of safety?

Answer 16: No. We not only accept but fully agree that no all non-conformances are equal. Some require immediate shut down, others do not. But longstanding reliance on fire watches in lieu of compliance with fire protection regulations seems far beyond the pale. Then NRC Chairman Ivan Selin testified to the Congress in 1993 that fire watches were an acceptable compensatory measure for up to six months. Yet fire watches remain in force nearly 15 years later. Fire watches have essentially become the de facto regulations, all without public notice and comment.

Question 17: For UCS, have you considered that the NRC rulemaking process is what it is, in part because UCS has sued the NRC, and the result is the slow cumbersome approach? In short, the process is designed to prepare to be sued by you.

Answer 17: UCS has successfully sued NRC in the past - the backfit rule and the Sholley amendment process lead the list. We accept that these suits likely delayed subsequent rulemakings. But if the factors causing delays were honestly allocated, UCS's share would be miniscule compared to those caused by NRC's ineffectiveness. So, we'll plead guilty to misdemeanor delay if NRC pleads guilty to felony delay.

Question 18: Would development of guidance documents help developers of rule to find issues with rule before it is promulgated?

Answer 18: The language for any rule goes through an independent vetting process before it is promulgated. Guidance development while the rule language is going through this vetting process can only help in the issuance of a final and better defined guidance document.

Question 19: If the technical basis is developed and approved, then why does final rulemaking still take many years? If technical basis is correct, then improvements in safety are unnecessarily delayed.

Answer 19: The development of a Technical Basis before initiating a proposed rule creates a stable environment instead of hindrance to the rulemaking process and can result in decreasing scheduling deadlines. This document does not guarantee shortening the schedule of the final rule because other constraints may play a larger role in determining the schedule. The Administrative Procedures Act (APA) is the federal law that establishes the administrative requirements for issuance all federal regulations. The APA requires a public notice and comment process intended to facilitate public stakeholder participation in the rulemaking process. There other numerous other federal laws (Paperwork Reduction Act, National Environmental Protection Act, Small Business Regulatory Enforcement Act, Regulatory Effectiveness Act, etc.) and executive orders that impose additional requirements on rulemaking. The overall effect of these regulations is to make federal rulemaking a careful and deliberate process and ensure that federal regulations are not issued without adequate opportunity for input by the regulated parties and other interested public stakeholders.

Question 20: What is the relationship on timing of NRC rulemaking, generic letters, or NUREGs once they are issued or proposed in the U.S. before it appears in Japan's structure?

Answer 20: The Japanese regulator always gather the latest regulatory information from the U.S. and other countries in order to enhance the effectiveness of regulation in Japan. Aircraft impact rulemaking is one example. NISA, the Japanese regulator, already modified the regulation on it. Since the U.S. and Japan both have own jurisdiction, there is no legal relationship on timing of the NRC rulemaking, generic letters, or NUREGs between two countries. But, exchanging the information is quite essential, I believe.

Question 21: UCS has made comments about the NRC's procedures for a number of years. Does the NRC believe any of his comments are valid? If so, has the NRC ever adapted any?

Answer 21: From UCS's perspective, our "ranting" has led to constructive changes to NRC's procedures. For example, the RIC used to require a fee to attend. UCS refused to participate, pointing out that the fee could easily be paid by industry representatives but presented an obstacle for the public. The NRC agreed and eliminated the fees. As another example, UCS pointed out that members of the public attending NRC's "public" meetings were not allowed to ask questions or provide comments. The NRC revised its public meeting process to the current format of three meeting types with varying levels of public participation.

Question 22: What is Japan's opinion of the NRC rulemaking process?

Answer 22: I would like to express my opinion instead of Japan's opinion. I think it is significant that the NRC staff people pay attention to the comments from the public in rulemaking process. Concerning the length of rulemaking process, it is not appropriate to compare between two countries without considering the difference in involving stakeholders in rulemaking process, I think.

Question 23: Mr. Yamashita, do the technical people write your regulations with the help of your attorneys OR do the attorneys write your regulations with the help of your technical staff?

Answer 23: Usually, the technical people draft the text and then the legal people re-write such draft in order to make it clearer. But sometimes, when we are in a hurry, the legal people listen from the technical people what and how the technical people want to regulate and then the legal people start to write the text by themselves from the beginning. I would like to emphasize that writing the text is the cooperative work between the technical people and the legal people. And I think it is essential for the technical people to understand that how their wants have been materialized as legal

		text, because when we consider the process of enforcement, the technical people are required to enforce the regulation in an appropriate way. If they don't know the meaning or they don't interpret appropriately, it might cause a problem. Therefore, as Director for Legislative Affairs, I intentionally encouraged the technical people to read and interpret the text of the regulation in their daily regulatory activities. **Session Chair:** Michael Case, NRC/NRR **Session POC:** Paulette Torres, NRC/NRR, tel: (301) 415-5656 e-mail: PAT3@nrc.gov
10:00 am-11:30 am Brookside		**Collaboratively Addressing PRA Challenges: Human Reliability Analysis, Fire Safety, and the Treatment of Uncertainties** Track 3 – Reactor Research All questions answered onsite. **Session Chair:** John Monninger, NRC/RES **Session POC:** Lauren Killian, NRC/RES, tel: (301) 415-0029 e-mail: LAK@nrc.gov
10:00 am-11:30 am Salon D		**New Reactor Design Reviews and Engineering Issues** Track 4 – New Reactors **Question 1:** Will NRC develop fundamental design competence (like Allen Bradley, Siemens, Honeywell) in order to validate & verify digital I&C SSC design performance requirements? **Answer 1:** The NRC doesn't plan to develop competence with a particular product or vendor of I&C equipment. Rather, we develop competence in technical areas such as software development, data communications, etc. We will sometimes utilize training from a vendor or on a particular product to gain familiarity with that product and others similar to it. **Question 2:** Why doesn't NRC consider "upgrade" as part of I&C development lifecycle? **Answer 2:** The retirement phase of the I&C lifecycle was not depicted in the presentation. Digital upgrades would be considered as part of this phase. The Office of Nuclear Reactor Regulation (NRR) addresses digital upgrades for safety-related I&C systems. **Question 3:** At what point in the DAC scheduling process do you envision the vendors or COL applicants having their overall certification or licensing schedule impacted due to missed milestones based upon non-submittal or low quality submittal of design acceptance documents? **Answer 3:** The NRC is able to make a final safety conclusion in a design certification or licensing review based on adequate design acceptance criteria (DAC). Therefore, the

Question 4: The NRC has approved the use of the incoherency model to provide a more realistic evaluation of the effects of high frequency ground motion at sites. Why was this not shown in your graphs or discussed?

Answer 4: The presentation made reference to permitting the use of advanced analytical methods for seismic analysis which includes consideration of incoherency effects. Slide 7 of the presentation was an example of the reduction in spectral amplitude that resulted from considering incoherency in ground motion.

Question 5: An agreement has been reached with the staff at a Feb 13th public meeting on the required analysis to address high frequency effects. Why this is considered an open item?

Answer 5: The NRC staff is currently assessing the implementation of the interim staff guidance on high frequency issues in the Design Certification applications. The staff anticipates a quick finalization of the interim staff guidance based on the common understanding between the industry and the NRC staff. The NRC expects to post the draft on the website in mid -April and issue the final ISG approximately one month later.

Question 6: Please define "Incoherence"

Answer 6: Earthquakes give rise to random spatial variation of ground motions. This characteristic is referred to as incoherency of ground motions. Use of incoherency in seismic analysis affects the spectral response of structures. The larger the foundation footprint of the structure the larger is this effect.

Question 7: What is the safety concern related to high frequency motion?

Answer 7: The high frequency zone is the range of frequencies beyond 10Hz. All plant sites do not have site responses that have significant energy content in the high frequency zone. The certification of a specific design is based on the seismic design response spectra selected by the design center. If the site specific spectra exceed the design response spectra, further evaluation of the structures, systems, and components affected by the exceedance is needed to ensure their availability during a safe shutdown earthquake. In some of the Central/Eastern United States rock sites this ground motion exceedance has been in the high frequency range.

Question 8: Can specific site data acquisition (e.g. borehole and shear wave velocity data acquisition) mitigate the burden of generic eastern United States (US) prediction of higher ground motion at frequencies > 10 Hz?

Answer 8: Guidance for obtaining geological, seismological and geotechnical engineering data acquisition is specified in Standard Review Plan Section 2.5. Based on the site data and detailed evaluation, each Combined License applicant develops their site specific Ground Motion Response Spectra. Some of these site specific ground motion spectra have shown higher ground motion at frequencies greater than 10Hz when compared to the seismic design response spectra selected by the standard design.

Question 9: Design Response Spectra (DRS) are currently based on RG 1.60 which may be non conservative with new design response spectra at high frequency. Does NRC plan to revise RG1.60? If yes when?

Answer 9: The NRC does not have any plans for revising Regulatory Guide RG 1.60. The high frequency zone is the range of frequencies beyond 10Hz. All new plant sites do not have site responses that have significant energy content in the high frequency zone. For those sites RG 1.60 spectral shape is appropriate. Under the current licensing process the applicant is required to confirm that site specific ground response

		spectra will be bounded by the seismic design response spectra selected by the standard design. Each standard design may establish their design basis response spectra using a combination of RG 1.60 spectra and spectra representative of sites with high frequency content. This allows for the standard design to be used at different sites. **Question 10:** RG 1.61 damping is based on RG 1.60 seismic (spectra). With higher frequency response spectra, the damping will be smaller. That means RG 1.61 will be non conservative with CSDRS **Answer 10:** While no specific question was asked, we are providing the following clarification: Damping is a measure of the energy dissipation capacity of a system as it responds to dynamic excitation. RG 1.61 is not based on RG 1.60 spectra but recommends acceptable damping values for different types of construction at different levels of earthquake. **Question 11:** I have been told that none of the standard design being certified will work for sites in western part of United States. Curious as to what is the difference/challenge? **Answer 11:** Some of the standard designs have utilized a combination of RG 1.60 spectra and spectra representative of sites with high frequency content as their design response spectra. RG 1.60 is predominantly based on earthquake recordings from the Western United States. The standard design response spectra are not site specific. Under the current licensing process the applicant is required to confirm that site specific ground response spectra will be bounded by the seismic design response spectra selected by the standard design. **Question 12:** Would it be possible to use a seismic PRA to help license a plant with high frequency exceedance? If risk is very, very low, is it OK? **Answer 12:** The seismic probabilistic risk assessment would only confirm the robustness of the seismic design. There is a minimum regulatory design requirement, in Appendix S to Code of Federal Regulations, Title 10, Part 50 that must be met without consideration of risk. Currently, the industry has not proposed using probabilistic risk assessment techniques to determine contributions from high frequency sensitive equipment and components to risk. **Session Chair:** Laura Dudes, NRC/NRO **Session POC:** Denise McGovern, NRC/NRO, tel: (301) 415-0681 e-mail: DLM7@nrc.gov
10:00 am-11:30 am Lower Level Entrance Marinelli & Executive Blvd		**Incident Response Experience** Track 5 – Nuclear Security, Emergency Preparedness, Fuel Cycle All questions answered onsite. **Session Chair:** Brian McDermott, NRC/NSIR **Session POC:** Janelle Jessie, NRC/NSIR, tel: (301) 415-6775 e-mail: JRB6@nrc.gov
10:00 am-11:30 am Salon F-H		**New and Advanced Reactor Research** Track 3 – Reactor Research **Question 1:** What are some of the possible environmental issues associated with the PBMR and have there been resources allotted to resolve these issues? **Question 2:** Is there any plan to run tests at both high temperature and high pressure to

		get HT correlations under prototypic conditions? **Session Chair:** Christiana Lui, NRC/RES **Session POC:** Lauren Gibson, NRC/RES, tel: (301) 415-0114 e-mail: LKG1@nrc.gov
10:00 am – 4:00 pm White Flint Amphitheater		**Post-Approval Site Inspection for License Renewal – Part I & II** Track 2 – Operating Reactors **Question 1:** What will the inspection report look like for the IP 71003 inspection; will it be similar to that issued for the 71002 inspections? **Question 2:** Can the facilities docket the technical details of the review of newly identified SSCs in separate correspondence/packaging/submittals in accordance with 50.59 and 50.71 requirements? If so, the licensees could then provide a comparable level of detail as in their other systems or SSCs that are existing plant UFSAR. **Session Chair:** P.T. Kuo, NRC/NRR **Session POC:** Ngoc (Tommy) Le, NRC/NRR, tel: (301) 415-1458 e-mail: NBL@nrc.gov