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

RIC 2015 March 10–12
27TH ANNUAL REGULATORY INFORMATION CONFERENCE

ANSWERS TO UNANSWERED QUESTIONS FOR RIC 2015

Tuesday, March 10, 2015

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TECHNICAL SESSIONS
Tuesday, March 10, 2015, 1:30 p.m. – 3:00 p.m.

T1 A Review of Public Participation in Nuclear Regulatory Proceedings in the U.S. and International Alternatives

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

Session Coordinator: Twana Ellis, Program Analyst, Program Support and Analysis Staff, ASLBP/NRC, 301-415-7703, Twana.Ellis@nrc.gov

The questions below were not answered during the above session.

Question 1 [addressed to Jean-Christophe Niel, ASN]: When you mentioned the Aarhaus Convention, you indicated the second pillar allowed public participation in the decision. Could you please clarify the nature of this participation?

Answer 1 [response from Jean-Christophe Niel, ASN]: There are several processes for the participation of the public:

- public debate to discuss the principle of the construction of large infrastructure including NPP or waste disposal site (this process is also applicable to non-nuclear construction as dam, industrial facility…);

- public enquiry to discuss of a detailed project;

The presentation shows the different ways to deal with public participation. Part of these go through intermediates (NGO, local committee for information known as CLI, etc.)

The public can interact directly through the various processes projected briefly in the presentation (public debate, public enquiries, public participation, and impact studies). In this case, their comments are taken into consideration and the way they are taken into account is explained.

The second pillar of the Aarhaus Convention is “the right to participate in environmental decision-making. Arrangements are to be made by public authorities to enable the public affected and environmental non-governmental organisations to comment on, for example, proposals for projects affecting the environment, or plans and programmes relating to the environment, these comments to be taken into due account in decision-making, and information to be provided on the final decisions and the reasons for it” ("public participation in environmental decision-making");

Question 2 [addressed to Jean-Christophe Niel, ASN]: In the NRC process, the NRC Staff frequently participates and takes the same position as the applicant. Is there a similar entity in the French system? If yes, do you find their participation helpful?

Answer 2 [response from Jean-Christophe Niel, ASN]:

1. If ASN is not convinced by operator arguments and safety demonstration, the authorization is not given
2. The participation of the public is definitely helpful
    - During my presentation, I suggest that it contributes to accountability of the safely authority and to its independence
    - Public participation may lead to modify the content of the decision
3. There is nevertheless an intrinsic difficulty to have a very broad involvement of the public
    - Topics are very often technical and complex
    - Very skilled people on these topics outside of the operators and safety authority that are ready to spend time on these topics are not so numerous
    - It is an obstacle to larger public participation
        - o We should try to deal with this, for example by motivating third parties to get involved in this field;

Question 3 [addressed to Jean-Christophe Niel, ASN]: How do you manage (deal with) the need of information for the public X the need of protecting the know-how of civil nuclear industry (and investments done for some decades) and which is exposed for licensing?

Answer 3 [response from Jean-Christophe Niel, ASN]:

1. The laws and regulations give free access to information related to nuclear safety to anyone. This information can be detained by public authorities or operator. It introduces exceptions which have to be justified.
2. Reference to protection of know-how must not be a way to avoid giving the relevant information.
3. ASN considers it a part of its job to check, as far as possible, that protection of know-how is not overused.
4. In the end, the issue can be arbitrated by justice.
5. My experience in ASN is that safety files, necessary to safety authority, rarely go into technical details that deserve proprietary information protection.

Question 4 [addressed to Jean-Christophe Niel, ASN]: Is ASN opened to public including international people? Are the results of stress tests available to access to see on the ASN website? What kind of information is prohibited to open to the public?

Answer 4 [response from Jean-Christophe Niel, ASN]: The various laws that deal with transparency do not only apply to French citizens but also to foreign ones. Lots of documents related to stress tests are already available on ASN's website (specifications, licensees' reports, ASN' decisions about related requirements). Most of these documents are also available in English. ASN can also provide Answers to specific Questions asked by stakeholders and provide documents which are not already available on ASN's website.

Question 5 [addressed to Petteri Tiippana, STUK]: Has Finland ever made a decision against an applicant – either in a decision in principle phase or afterwards?

Answer 5 [response from Petteri Tiippana, STUK]: Yes, twice. Parliament rejected Government's Decision in Principle in 1993. And in 2010 Government rejected an application for a Decision in Principle.

Question 6 [addressed to Petteri Tiippana, STUK]: How does STUK take into account safety comments from public that are not the focus of environmental impact assessment or decision in principle?

Answer 6 [response from Petteri Tiippana, STUK]: Public has possibility to submit their concerns on nuclear safety or on STUK's activities/decisions either directly to STUK or via the ministries or directly to the Parliament's ombudsman. STUK reviews all Questions and comments and provides an Answer to those.

Question 7 [addressed to Mark Leblanc, CNSC]: What is the relationship of the consent-based siting process for a geologic repository and the intervention process of your hearings? Does the contested hearing process undermine a consent-based siting process for a repository?

Answer 7 [response from Mark Leblanc, CNSC]: In Canada, the Nuclear Waste Management Corporation is conducting a siting process to select 1 or 2 sites where the population would welcome/accept the construction of a fuel waste deep geologic repository. This is coordinated separately from the activities of the CNSC, which remains responsible for the licensing of the facility based only on safety considerations, and without regard to the fact there is a consenting community. So, these are 2 distinct processes, each serving their separate purposes.

Question 8 [addressed to Mark Leblanc, CNSC]: Please expand on Code of Behavior. What is expected or not tolerated by members of the public?

Answer 8 [response from Mark Leblanc, CNSC]: The Code of Conduct is included on my presentation slides, available on the RIC2015 website. Please phone at 615-858-8058 or email at marc.leblanc@cnsc-ccsn.gc.ca if you wish to discuss. It is posted in the hearing room, and is sent to all public hearing and meeting participants prior to the proceedings.

Question 9 [addressed to Mark Leblanc, CNSC]: What is the annual average amount of public participation? What is the amount that the government pays to have the public participate?

Answer 9 [response from Mark Leblanc, CNSC]: About 200 to 300 participants per year, including non-governmental organizations, members of the public, unions, municipalities, etc. For example, a hearing is planned in Kincardine (Ontario) for the renewal of a nuclear power plant license. The Commission has received 144 interventions (60 will make oral presentations and 84 will limit their participation to written submissions). Participant funding was provided to 10 requesters (the total amount received is less than $100K). The amount available per year is approximately $900K.

Question 10 [addressed to Mark Leblanc, CNSC]: In light of the fact that your commissioners are part-time, have any concerns arisen related to potential conflicts of interest between the Commissioners, work as regulators and whatever work they do with the rest of their time?

Answer 10 [response from Mark Leblanc, CNSC]: The CNSC and its commissioners are very vigilant in terms of ensuring there are no real or perceived conflicts of interest. As most commissioners do not come from the nuclear field – but rather from other areas of endeavors – this has not been a real issue. In the few cases where there could have been a perceived conflict, the commissioners have excused themselves from that proceeding.

Question 11 [addressed to James Glasgow, Pillsbury Winthrop Shaw Pittman, LLP]: How would the administration's Blue Ribbon Commission consent based siting approach be impacted by the NRC's contested hearing process? Does a contested hearing undermine a consent-based siting process?

Answer 11 [response from James Glasgow, Pillsbury Winthrop Shaw Pittman, LLP]: This question may best be addressed within the context of the RIC session in which it was raised. In that session, the speakers discussed the manner in which the nuclear regulatory bodies of several countries allow public participation with respect to environmental assessments and licensing proceedings for nuclear power stations. The consistency of the NRC's rules regarding intervention in NRC hearings with a consent-based siting process in the U.S. may be considered in light of the process employed by other countries that have provided significant opportunities for members of the public to participate in the establishment of national policies and laws concerning long-term storage and disposition of used nuclear fuel. For example, Canada established the Nuclear Waste Management Organization (NWMO) to develop "collaboratively with Canadians a management approach for the long-term care of Canada's used nuclear fuel." The manner in which the public participated in this Canadian initiative was discussed in a November 2005 report by NWMO, entitled "Choosing a Way Forward: the Future Management of Canada's Used Nuclear Fuel." Various means of obtaining the "consent" of the public were also employed by the Governments of Finland and Sweden in their consideration and licensing of geologic repositories for used nuclear fuel (see e.g. presentations during WM 2014, organized by WM Symposia Inc.).

Question 12 [addressed to James Glasgow, Pillsbury Winthrop Shaw Pittman, LLP]: What are some of the best ways to allow for public participation in hearings without unduly burdening utilities with the public espousing anti-nuclear sentiments?

Answer 12 [response from James Glasgow, Pillsbury Winthrop Shaw Pittman, LLP]: The NRC's rules regarding public participation in NRC licensing proceedings (10 CFR section 2.309) require that persons who file petitions to intervene must submit at least one admissible contention and show that they have a legally sufficient interest to establish "standing." NRC's requirements regarding standing are in accordance with judicial concepts of standing to sue. Section 189a of the Atomic Energy Act does not provide an unqualified right to a hearing. The Commission has repeatedly observed that the NRC is authorized to condition that right by establishing reasonable procedural requirements. The NRC thus does not grant intervenor status to members of the public who merely wish to espouse anti-nuclear sentiments. Persons who wish to submit comments to the NRC in connection with licensing proceedings for nuclear power stations and other nuclear facilities may do so, of course, without becoming intervenors.

Question 13 [addressed to all]: What is the major obstacle in establishing public trust that your agency has encountered?

Answer 13 [response from Jean-Christophe Niel, ASN]: Even with evolutions which took place during past years (especially the law on security and transparency in nuclear matters known as the TSN Act) a suspicion remains in the general public that issues are still hidden from them. Anyhow, to gain trust from the general public is a lengthy process that could take decades. On the contrary, losing this trust is very easy.

Answer 13 [response from Min-Tsang Chang, AEC]: The anti-nuclear voice is louder than the nuclear supporter. Many people are reluctant to trust the government regarding nuclear issues.

Answer 13 [response from Petteri Tiippana, STUK]: This is difficult to say since public shows fairly little interest on nuclear and radiation safety issues in Finland. Lately, challenging areas have been related to gaining trust on the safety evaluation on the final disposal of spent fuel (deep geological repository), and on the safety on the use of mobile phones and wireless technology (non ionization radiation). Major obstacle is to get people interested in radiation and nuclear safety matters.

Answer 13 [response from Marc Leblanc, CNSC]: There are very polarized views when it comes to nuclear. While the CNSC makes considerable efforts to disseminate objective scientific information about nuclear safety, there are pockets of the population that see this as nuclear promotion. The CNSC continues to be as transparent as possible by conducting and webcasting its public hearings, allowing public participation, disseminating scientific information, conducting CNSC 101 information session in affected communities, releasing nuclear safety videos on its website and Youtube, maintaining a very comprehensive website, etc.

Question 14 [addressed to all]: What opportunities exist for members of the public to challenge the actions of nuclear authorities?

Answer 14 [response from Jean-Christophe Niel, ASN]: Any decision from ASN can be challenged by the State Council (Conseil d'Etat) which is the highest administrative court in France. Any French citizen can make a request to this court. The whole process is strictly defined in the law.

Answer 14 [response from Min-Tsang Chang, AEC]: Hearings must be completed before licensing at all radioactive waste facilities. The final site of radioactive waste disposal has to be determined by the local referendum. Furthermore, the people in our country have the right of lodging complaints, and instituting legal proceedings.

Answer 15 [response from Petteri Tiippana, STUK]: Decisions and statements from the regulators are mostly public in Finland (except for security and commercial issues). Most significant decisions and statements are published and also public press conferences are organized. Public can challenge authorities actions and decisions by appealing to the Parliament's ombudsman.

Answer 15 [response from Marc Leblanc, CNSC]: In Canada, the legislation governing the CNSC provides for public participation in licensing hearings of major nuclear facilities. As licenses are typically issued for 5-year periods, this provides frequent opportunities for members of the public to express their views. In addition, the public is invited to provide submissions in the context of the annual reports on industry segments such as the nuclear power plants, the uranium mines and mills, the fuel fabrication and research facilities, etc. Also, there are opportunities to comment on all new or amended regulatory documents and industry standards.

Decisions of the Commission can be challenged through judicial review applications to the Federal Court of Canada.

Question 16 [addressed to all]: Who has social scientists on their agency's staff?

Answer 16 [response from Jean-Christophe Niel, ASN]: At ASN, we have a network of people in charge of organizational and human factors which is animated and coordinated by a skilled specialist of this field. IRSN, ASN's TSO, has also a unit in charge of this topic.

Answer 16 [response from Min-Tsang Chang, AEC]: In the AEC, a specific division is appointed to take care of social matters. Social science scholars and scientists are involved in the research projects every year.

Answer 16 [response from Petteri Tiippana, STUK]: STUK has and has had (psychologists and social psychologists).

Question 17 [addressed to all]: No one has mentioned security. Given the overlap of safety and security (e.g. cyber), to what extent does the regulator ensure operators follow AEA nuclear security guidelines or other security imperatives? Is force-on-force testing done or other red-team testing (e.g. cyber)?

Answer 17 [response from Jean-Christophe Niel, ASN]: ASN is not in charge of issues related to security or physical protection. ASN has exchanges on a regular basis with the authority in charge of security issues.

Answer 17 [response from Min-Tsang Chang, AEC]: The AEC requests Taiwan's NPP operators to refer to the requirement of NRC RG 5.71.

Answer 17 [response from Petteri Tiippana, STUK]: In Finland operators have to follow Finnish regulations and regulatory guides. IAEA safety standards and guidelines are used as a reference when Finnish regulations and guidelines are established and developed. Inspections and oversight are used to ensure compliance with requirements.

Answer 17 [response from Marc Leblanc, CNSC]: The Canadian Nuclear Safety Commission gives a lot of attention to the matters of nuclear safety and security. It has a comprehensive dedicated team whose responsibility is nuclear security, and compliance with Canadian and international security requirements. The CNSC works closely with the IAEA and the NEA in this regard. CNSC conducts security inspections at key nuclear facilities in Canada, often integrated with safety inspections. The CNSC also participates in International Physical Protection Advisory Services missions, under the auspices of the IAEA. The International Physical Protection Advisory Service (IPPAS) was created by the IAEA to assist States in strengthening their national nuclear security regime. IPPAS provides peer advice on implementing international instruments, and IAEA guidance on the protection of nuclear and other radioactive material and associated facilities.

Question 18 [addressed to all]: After the highest authority in your country (such as Parliament) makes a decision to go ahead with a project, can they change their decision?

Answer 18 [response from Jean-Christophe Niel, ASN]: In France, there is a hierarchy among official texts. The constitution is the highest level and is above a law. A law, voted by the Parliament, is above a decree… A text of a given level can lead to the modification of a text of the same level as soon as it respects or does raise any contradiction with higher level texts (as an example, a new law can lead to the modification of an existing law or to its abrogation, but it has to be ensured that it respects the Constitution).

Answer 18 [response from Min-Tsang Chang, AEC]: Yes, the highest authority can change their decision.

Answer 18 [response from Petteri Tiippana, STUK]: No, they cannot. Finland has three step licensing. Parliament is involved only in the first step (Decision in Principle), where the Parliament either ratify or reject Governments Decision in Principle (which is most of all about whether using nuclear energy is for the overall good of the society). There is no possibility to appeal on Parliament's decision. If the Decision in principle is favorable to the project, Government later on makes licensing decisions (construction license and operating license) and in those steps Government again has a possibility to re-evaluate if giving the license is for the overall good of the society. Government's decisions can be appealed to the administrative court.

Answer 18 [response from Marc Leblanc, CNSC]: Yes.

T2 Enhancing Nuclear Safety and Security Practices through International Peer Review Missions

Session Chair: David Skeen, Deputy Director, Office of International Programs, NRC, 301-415-2344, David.Skeen@nrc.gov

Session Coordinator: Emily Larson, International Relations Specialist, OIP/NRC, 301-415-1151, Emily.Larson@nrc.gov

The questions below were not answered during the above session.

Question 1 [addressed to Vesselina Ranguelova, IAEA]: How does the IAEA prioritize the OSART mission schedule? Is the schedule available on the IAEA website?

Answer 1 [response from addressed to Vesselina Ranguelova, IAEA]: The IAEA performs 6-8 OSART mission per year at the request of the IAEA Member States (MSs). The IAEA has different arrangements with different countries as to which missions are requested and will be performed. The IAEA has conducted 182 OSART missions since 1982. Most recent OSART missions and requested missions in each country are available at:

http://www-ns.iaea.org/downloads/ni/s-reviews/osart/table-of-countries.pdf

Question 2 [addressed to Vesselina Ranguelova, IAEA]: Before conducting an OSART mission or permission, does the team review the insights from any probabilistic safety assessment or peer review, is such things exist?

Answer 2 [response from Vesselina Ranguelova, IAEA]: The IAEA OSART team reviews prior to the mission the Advanced Information Package prepared by the plant. This document contains a lot of technical information, including insights from safety assessments and peer reviews. In addition, in preparation for the mission, the IAEA OSART team uses information on international operational experience relevant to the type of reactor to be reviewed. The development of PSA models and possible use of any PSA applications at the plant is reviewed during the OSAR mission itself.

Question 3 [addressed to Vesselina Ranguelova, IAEA]: How are best practices reconciled with national cultural constrains such as: politically opaque cultures; national pride and sovereignty issues; no nuclear history or experience; or inexperienced regulators?

Answer 3 [response from Vesselina Ranguelova, IAEA]: The IAEA OSART mission has several modules which are considering those issues: Leadership and Management for Safety and Interactions of Human, Technology and Organization. The IAEA Safety Standards are used as a basis for judgments. The IAEA Safety Standards are available at: http://www-ns.iaea.org/standards/default.asp?s=11&l=90

Question 4 [addressed to Vesselina Ranguelova, IAEA]: What criteria and measures are used to determine with a plant is operating safely? From you experience reviewing nuclear plants in different countries, what factors lead to differences in safety performance at plants in different countries?

Answer 4 [response from Vesselina Ranguelova, IAEA]: During the OSART missions the IAEA Safety Standards are used as a basis for judgments. The IAEA Safety Standards are available at: http://www-ns.iaea.org/standards/default.asp?s=11&l=90

The factors influencing safety performance in different countries are various, including availability of resources, prescriptive or non-prescriptive regulatory regime; strong or not safety structure, access to international operational experience, etc.

Question 5 [addressed to Jacques Regaldo, WANO]: Why is the WANO peer review missions not published publicly?

Answer 5 [response from Jacques Regaldo, WANO]: WANO was founded after Tchernobyl accident to promote and foster transparent exchanges of information among nuclear operators to enhance nuclear safety worldwide. We ensure our members a strict confidentiality of the results of our programmes to maintain the highest possible trust among nuclear operators, and thus transparent exchanges of information.

Question 6 [addressed to Jacques Regaldo, WANO]: How are best practices reconciled with national cultural constrains such as: politically opaque cultures, national pride and sovereignty issues, no nuclear history or experience or inexperienced regulators?

Answer 6 [response from Jacques Regaldo, WANO: Constraints exist everywhere in various forms but WANO is based on one main principle: we are stronger together. It is true for new entrants that "cannot afford to make all the mistakes the older one did" to acquire experience, but it is also true for the older ones. Today all civil nuclear operators are members of WANO and doing so, they adopt this principle. Fukushima showed that an accident anywhere could impact everyone, and I believe our members see their appurtenance to WANO more as a chance than as a constraint.

Question 7 [addressed to Jacques Regaldo, WANO]: What criteria and measures are used to determine with a plant is operating safely? From you experience reviewing nuclear plants in different countries, what factors lead to differences in safety performance at plants in different countries?

Answer 7 [response from Jacques Regaldo, WANO]: WANO uses different kind of criteria and inputs to build its appreciation on the safety level of the plants. Operational KPIs that have been discussed and agreed among the four WANO regional centres and which are very similar to those used by INPO are used. The operating events reported by the plants on regular basis are also very useful. But WANO maintains regular contacts with its members and all available information is used to determine the safety level of the plants. This includes observations done during WANO peer reviews themselves based on safety performance indicators and criteria and observed best practices worldwide.

There are many factors that can lead to differences in safety performance but probably the most important on a long term perspective is the lack of openness and self-questioning attitude that some operators may have, because as soon as a person or an organisation considers he/she/it has nothing to learn from the others, a danger occurs.

T3 Environmental Health Physics: Risk Communication and the Use of Dose Assessment for Operating and Decommissioning Reactor Sites

Session Chair: Rebecca Tadesse, Branch Chief, Division of Systems Analysis, RES/NRC, 301-415-7490, Rebecca.Tadesse@nrc.gov

Session Coordinators: Katie Tapp, Health Physicist, DSA/RES, 301-251-7520, Katherine.Tapp@nrc.gov

Lisa Ramirez, Health Physicist, DSA/RES, 301-251-7546, Lisa.Ramirez@nrc.gov

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

T4 Improving the Way We Do Business – for Large Lights and Small Modulars

Session Chair: Stephen Koenick, Senior Policy Analyst, Division of Advanced Reactors and Rulemaking, NRO/NRC, 301-415-6631, Stephen.Koenick@nrc.gov

Session Coordinator: Dennis Galvin, Project Manager, Division of Advanced Reactors and Rulemaking, NRO/NRC, 301-415-6256, Dennis.Galvin@nrc.gov

The questions below were not answered during the above session.

Question 1: What are three lessons learned from the design-specific review standard (DSRS) process?

Answer 1 [response from Stephen Koenick, NRC]: The NRC has learned several lessons from the DSRS process:

1. It is important that the design is adequately developed by the prospective applicant such that the applicability of existing SRP sections can be determined; positions can be developed on key technical and regulatory issues; and design specific guidance can be developed.
2. While being "design-specific," there is a considerable portion of guidance from the mPower DSRS that is applicable to other small modular reactor (SMR) designs such as NuScale, because of the similarity of systems.
3. The development process for the DSRS enabled the NRC staff to identify a process to risk-inform certain review areas as discussed in the introduction to NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," (hereafter referred to as the SRP) "Introduction - Part 2: Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: Small Modular Reactor Edition," Rev 0, dated January 2014. The NRC staff will assess the effectiveness of the risk-informed process after implementation of the first SMR design certification application review.

SRP Introduction - Part 2, is available at the NRC's public website - NUREG-0800 Introduction Part 2

Question 2: What is the NRC doing to alleviate the Tier 2* burden associated with non-safety significant issues identified at Vogtle and Summer? A near-term solution is needed.

Answer 2 [response from Stephen Koenick, NRC]: Licensees can request to redesignate specific Tier 2* information as Tier 2 information, recognizing that the Tier 2* designation also encompasses the concept of design center standardization as well as the safety significance concept within the certified design information. This process is pursuant to 10 CFR 52.9(c) and in accordance with 10 CFR 50.90 for requests for license amendment and exemption. For example, by letter dated September 2, 2014, the NRC granted portions of Tier 2* Human Factors Verification and Validation Technical Reports listed in the UFSAR to be reclassified as Tier 2 information. This letter is available in the NRC Agencywide Documents Access and Management System (ADAMS) under Accession No. ML14073A752.

In addition, the NRC staff recognized that the clarity of Tier 2* information could be enhanced in Lesson 1 of the Post-Combined License Part 52 Implementation Lessons Learned Report, dated July 22, 2013 (ADAMS Accession No. ML13196A403). The NRC staff documented action plans to address the lessons in a memorandum, "Status of Action Plans in Response to the Post Combined License Part 52 Implementation Lessons Learned Report," dated March 7, 2014 (ADAMS Accession No. ML13357A259). The NRC staff held three meetings with the public regarding Tier 2* on May 8, 2014, October 6, 2014, and November 13, 2014. The meeting summary for the latest public meeting is available under ADAMS Accession No. ML15013A021. The NRC staff is currently considering objective and measurable criteria for future design information to assure that information is placed in appropriate tiers.

Question 3 [addressed to NRC]: What would be needed to reduce a US COL application review to a 24-month duration? What is the nuclear industry's greatest weakness in meeting COL/DCD schedule?

Answer 3 [response from Stephen Koenick, NRC]: The initial planning assumption for a COL application was 30 months; however, the NRC establishes the review schedule after determining that an application is complete and technically sufficient to conduct the review in a predictable timeframe. The biggest challenge in meeting COL review schedules to date has been that they have been conducted in parallel with a standard design certification application review. The design certification review process had been planned to take 42 to 60 months to complete. In reality, they have taken even longer and become the critical path to completion of the COL application review process.

Question 4 [addressed to Tom Kevern, NRC]: How does the NRC plan to deal with an application for a COL following an application for an early site permit (ESP), but before the ESP is issued? Two applications would be before the Commission at the same time.

Answer 4 [response from Tom Kevern, NRC and Mark Notich, NRC]: § 52.26(c) states "An applicant for a construction permit or combined license may, at its own risk, reference in its application a site for which an early site permit application has been docketed but not granted." The NRC staff will interact with potential applicants on this topic in the future.

Question 5 [addressed to Tom Kevern, NRC]: Current licensees and regulators continue to have to deal with issues arising from differences in the interpretation of terms as they were understood during initial licensing, up to 30 or 40 years ago, versus modern interpretations. What is being done to address this in new licensing activities?

Answer 5 [response from Tom Kevern, NRC]: Standardization of reactor designs and licensing processes, as provided by 10 CFR Part 52 and promoted by the NRC, enhances safety and reduces interpretation challenges. The NRC promotes the standardization of applications to enhance the safety, reliability, and availability of nuclear power plants, and facilitate a predictable and consistent method for application review. The agency's design-centered review approach (DCRA) is a strategy based on industry standardization of COL applications referencing a particular reactor design. When such standardization is achieved, the NRC staff conducts one technical review for each reactor design issue and uses this one decision to support the decision on a design certification application and on multiple COL applications. The DCRA strategy was initially addressed in RIS 2006-06, "New Reactor Standardization Needed to Support the Design-Centered Licensing Review Approach," May 31, 2006 (ADAMS Accession No. ML053540251) and the strategy continue to be a subject of the annual RIS information requests.

In addition to standardization, the NRC staff believes that capturing the clarification of the differences in interpretations in durable (unless we say what "durable" guidance is, we should use another term) guidance will further enhance safety and reduce interpretation challenges. The revision to RG 1.206 represents one of these efforts and provides interested stakeholders a forum in which to identify their concerns.

Question 6 [addressed to Russell Bell, NEI]: What is your view on DC/COL projects with foreign designs – KHNP's APR1400, MHI US-APWR, etc…? How do they help the US nuclear industry?

Answer 6 [response from Russell Bell, NEI]: In an increasingly global industry, it is hard to neatly categorize designs as foreign versus domestic. U.S. generating companies like choices and will choose an NRC-certified technology based on a range of market/business factors.

Question 7 [addressed to Russell Bell, NEI]: What could NRC do to facilitate licensing of "non-LWR" advanced reactor designs, which have the potential to offer better safety and performance?

Answer 7 [response from Russell Bell, NEI]: In the near term, NRC should continue its development of general design criteria for non-LWR designs including high-temperature gas, liquid metal cooled, and molten-salt designs.

Question 8 [addressed to Patricia Milligan, NRC]: How will multiple SMRs collocated on the same site affect the EPZ [emergency planning zone]?

Answer 8 [response from Patricia Milligan, NRC]: The NRC staff identified multiple SMRs collocated on the same site, or "modularity", as an issue in SECY-11-0152 "Development of an Emergency Planning and Preparedness Framework for Small Modular Reactors", dated October 11, 2011 (ADAMS Accession No. ML112570439) and the NRC staff is currently working on this issue.

Question 9 [addressed to Patricia Milligan, NRC]: How will SMRs located on an existing site with operating large light water reactors affect the existing EPZ?

Answer 9 [response from Patricia Milligan, NRC]: SMRs would not be expected to change the existing EPZs for the large light water reactors, but this will be evaluated on a case-by-case basis where needed.

T5 Severe Accident Progression and Consequence Analysis in Support of Regulatory Decisionmaking in Light of the Fukushima Accident

Session Chair: Patricia Santiago, Branch Chief, Division of Systems Analysis, RES/NRC, 301-287-7982, Patricia.Santiago@nrc.gov

Session Coordinator: Shannon Thompson, Reactor Systems Engineer, Division of Systems Analysis, RES/NRC, 301-251-7685, Shannon.Thompson@nrc.gov

The questions below were not answered during the above session.

Question 1 [addressed to Jon Barr, NRC, and Rick Wachowiak, EPRI]: It seems that only effects of radioactivity are taken into account for health risks. It seems however that the stress from the accident and relocation could induce health problems. Does it seem proper to evaluate that?

Answer 1 [response from Jon Barr, NRC, and Rick Wachowiak, EPRI]: This is correct, only health effects due to radiation exposure have been included in the metrics presented. This is standard for the type of evaluation performed and is consistent with regulatory changes such as extended licensing and power uprates.

Question 2 [addressed to Hossein Esmaili, NRC, Jon Barr, NRC, and Rick Wachowiak, EPRI]: Probabilistic results assume that the frequencies of hazards greater than design basis are negligible. Is that really the case?

Answer 2 [addressed to Hossein Esmaili, NRC, Jon Barr, NRC, and Rick Wachowiak, EPRI]: This is not the case. The hazard frequencies are based on observed and calculated data from published risk analyses that are considered "best estimate".

T6 Treatment of Uncertainty in Risk-Informed Decisionmaking

Session Chair: Sunil Weerakkody, Branch Chief, Division of Risk Assessment, NRR/NRC, 301-415-2870, Sunil.Weerakkody@nrc.gov

Session Coordinator: Doug Copeland, Reliability and Risk Analyst, Division of Risk Assessment, NRR/NRC, 301-415-1246, Douglas.Copeland@nrc.gov

The questions below were not answered during the above session.

Question 1: What do you see as the distinctive roles in decisionmaking of the distinction between aleatory and epistemic uncertainty? Is the distinction still supported?

Answer 1 [response from Fernando Ferrante, NRC]: The distinction is still supported in NRC guidance on the topic. NRC's Regulatory Guide (RG) 1.174, "An Approach For Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-specific Changes To The Licensing Basis" discusses the characterization of aleatory and epistemic uncertainty and its use in decisionmaking. RG 1.174 indicates that, because of their natures, they must be treated differently when creating models of complex systems. As stated in RG 1.174, aleatory uncertainty is associated with events or phenomena being modeled that are characterized as occurring in a "random" or "stochastic" manner and probabilistic models are adopted to describe their occurrences. The epistemic uncertainty is associated with the analyst's confidence in the predictions of the PRA model itself and reflects the analyst's assessment of how well the PRA model represents the actual system being modeled.

For decisionmaking, it is important to develop an understanding of the impact of a specific assumption or choice of model on the predictions of the PRA. With respect to aleatory uncertainty for a component, for example, data can be collected and the aleatory uncertainty involved with this input can be treated explicitly and propagated through the PRA models. On the other hand, modeling assumptions may be more challenging to capture but are equally important to address in any PRA. The important aspect is to recognize the existence of these uncertainties and their potential impacts to risk-informed decisionmaking (e.g., can they be modeled or not, and how can they impact a decision with the insights they provide). As stated in RG 1.174, the impact of using alternative assumptions or models may be addressed by performing appropriate sensitivity studies or by using qualitative arguments, based on an understanding of the contributors to the results and how they are impacted by the change in assumptions or models. Hence, the impact of both aleatory and epistemic uncertainties in PRA is still supported and can play distinctive but critical roles for decisionmaking.

Question 2: If we measure risk results against "soft" guidelines, doesn't this result in decisionmaking uncertainty as different decisionmakers look at the results differently using subjective judgment?

Answer 2 [response from Fernando Ferrante, NRC]: Because decisionmakers will always be faced with uncertainties when making decisions, it is important to appropriately recognize and characterize them (whether these are treated explicitly or implicitly). In this regard, guidelines may be defined with “hard” thresholds (e.g., 1E-4/year for core damage frequency for an operating large commercial reactor in the US). The critical aspect is to understand that the resulting insights from a risk assessment will include ranges that may or may not exceed a specific threshold or may straddle multiple risk criteria, i.e., “soft” results. In this case, it is important to ask if the exceedance of a threshold is significant or not based on the underlying uncertainty involved. In a risk-informed process, some flexibility has to be provided to ensure the most appropriate decision can be made in lieu of the information available. An integrated risk-informed framework should allow for a more consistent implementation of this approach with some level of consistency such that decisionmakers are not overwhelmed with wide ranging diverse opinions from different inputs but, instead, focus on the drivers that can change a decision (including both quantitative and qualitative criteria). If implemented and documented accordingly, clear and transparent decisions can be reached with risk assessments even in the face of large uncertainties.

Question 3: How do you regulate to guidelines? Is it only in the eye of the beholder? If it’s a guideline, what exactly is acceptable? How do you communicate this to the public?

Answer 3 [response from Doug True, ERIN Engineering]: A number of current regulatory tools are guidelines. In fact, Reg. Guide 1.174 used the “quantitative acceptance guidelines” to describe the approach to interpreting PRA results.

It should not be in the eye of the beholder. That is not beneficial to the regulator, the licensee, or the public. However, we need to be clear on how these guidelines should be interpreted. This is where we can do better. We have not always made it clear what needs to be provided to decision-makers in order to interpret these guidelines. Revision 1 of NUREG-1855 is an important step in this direction. The work that the Uncertainty Working Group did in 2014 identifies additional steps that can be taken to clarify this.

We can communicate to the public that we have a clear, repeatable process for applying these guidelines.

Question 4: Any progress in treating the uncertainty associated with safety culture?

Answer 4 [response from Doug True, ERIN Engineering]: This remains a difficult aspect of quantitative decision-making and is a good example of why we rely on a risk-informed process that considers inputs beyond the PRA such as defense-in-depth and safety margins.

Question 5: Decisions on the applications are ultimately “yes” or “no.” Is this at odds with the fuzzy nature of risk that you’ve described? In other words, does PRA support: application decisions (the process) or safety (the goal)?

Answer 5 [response from Mary Drouin, NRC]: Was not attempting to imply that risk has a "fuzzy" nature. A PRA, as with any engineering analysis, there are uncertainties with the analysis. There are processes that are in place to address the uncertainties associated with engineering analyses; for example, defense-in-depth. The Question is how are the uncertainties associated with a PRA addressed. PRA supports the decision being considered in an application and part of that includes looking at how close the PRA results meet the acceptance guidelines (e.g., safety goals).

Question 6: Uncertainty of unknowns may not be quantitatively solved so how do we know it's limits? Can we rank uncertainty by using p=1 values, cut sets to find which uncertainties contribute most to risk? Can we use Monte Carlo or uncertainty distributions to assess risks?

Answer 6 [response from Mary Drouin, NRC]: Uncertainty analyses for PRAs do not address the unknown unknowns; these uncertainties are addressed via other means such as defense-in-depth. When looking at PRA uncertainties, there are the parametric uncertainties which are addressed by calculating the mean values for the quantification of the, for example, initiating event frequencies, basis event probabilities, core damage frequencies, large early release frequencies, etc. However, there are also model uncertainties. These uncertainties are addressed by first identifying which ones are relevant and key to the results. This is determined by evaluating how the PRA model is impacted by these uncertainties and then performing sensitivity analyses to determine their quantified effect on the actual quantified results.

Question 7: In the context of uncertainty, how is the effort to address the unknown unknowns or the consideration of the "Black Swan" reflected in the results; that is, beyond the known unknowns?

Answer 7 [response from Mary Drouin, NRC]: Uncertainty analyses for PRAs do not address the unknown unknowns; these uncertainties are addressed via other means such as defense-in-depth – assuring, for example, there are both prevention and mitigation measures, single failure criterion, safety margins – by continually asking the Question "what if this fails"?

Question 8: Can you describe how you manage unknown risk? An example would be helpful. How do you determine when unknown risk is too high?

Answer 8 [response from Homayoon Dezfuli, NASA]: Managing unknown or underappreciated (UU) risks involves application of a suite of heuristic system safety and systems engineering best practices whose objectives include:

- **Incorporate appropriate historically-informed defenses against UU risks into the design** — This involves taking measures to minimize the possibility that an incomplete understanding or analytical characterization of the system will result in a system that may exhibit undesirable safety performance. These measures go beyond those explicitly derived from the safety analysis, to include measures such as safety margin, failure tolerance, failsafing, and emergency operations, which have developed historically and are recognized as best practices in their engineering disciplines.

- **Maximize discovery of UU hazards during system realization/operation** — This involves strategies such as liberal instrumentation, monitoring and trending, post-flight anomaly investigation, application of modern safety analysis methods, Precursor Analysis, and testing. For example, new technology and new applications of existing technology are adequately tested within the larger system before becoming operational.
- **Minimize the introduction of potentially UU hazards during system realization/operation** — This relates to adherence to best practices in system realization and operation, and the imposition of engineering discipline to keep system operation within the design intent. The following rules of conduct are known to have a positive effect on reducing UU risks:
    - Realism in budgets and schedules.
    - Avoidance of unneeded complexity in the realization and operation of the system.
    - Management promotion of a safety culture in which information about safety risks is discussed openly and inclusively between levels of the organization involved in the realization and operation of the system.
    - Maintenance of management oversight of distributed sources and suppliers.
    - A process for communicating and correcting the deficiencies uncovered by inspections and audits.

The ability of a space program to manage UU risks is a function of the quality of efforts made (or strategies employed) to fulfill the above objectives. A determination that UU risk is high needs to be based on overall judgments of how the program is addressing the objectives and the soundness of the evidentiary basis for it. For additional Information, refer to NASA/SP-2010-580, the NASA System Safety Handbook, Volume 1 (http://www.hq.nasa.gov/office/codeq/doctree/NASASP2010580.pdf).

Question 9: What was the model used at the time of the Columbia spaceship disaster? How was the model improved (eliminating risk) at present time?

Answer 9 [response from Homayoon Dezfuli, NASA]: The report of the Columbia Accident Investigation Board (CAIB) (https://www.nasa.gov/columbia/home/CAIB_Vol1.html) contains an exhaustive analysis of the physical and organizational causes of the Columbia crash.

Post-Columbia, NASA has made significant improvements to its governance model and technical and decision making processes. (Refer to NPD-1000.0B, Governance and Strategic Management Handbook at http://nodis3.gsfc.nasa.gov/npg_img/N_PD_1000_000B_/N_PD_1000_000B_.pdf.) Examples of such improvements include:

- In conjunction with CAIB recommendation R7.5-1 and as a key part of NASA's overall system of checks and balances, the Agency has established the role of Technical Authority (TA) to provide independent oversight of all space flight programs and projects in support of safety and mission success. Additional information on the TA function can be found at http://fpd.gsfc.nasa.gov/NPR71205D/Technical_Authroity_FAQs.pdf.

- The Agency has made significant improvements to its risk management and system safety technical processes. System safety and mission reliability have been elevated to the status of system performance attributes on a par with technical, cost, and schedule performance. PRA techniques are now used for aggregation of individual risk issues to produce system-level safety and mission reliability performance measures (e.g., Probability of Loss of Crew (P(LOC)), Probability of Loss of Mission (P(LOM))). This makes possible the levying of, and managing to, system-level safety and mission reliability performance requirements for high-significance missions (e.g., human spaceflight)
- The Agency created the NASA Engineering and Safety Center (NESC) to perform value-added independent testing, analysis, and assessments of NASA's high-risk projects to ensure safety and mission success (see http://www.nasa.gov/offices/nesc/home/index.html).
- The Agency has established a multi-level safety reporting process for employees and NASA contractors to ensure that safety issues are promptly reported and considered independent of programmatic processes. This safety reporting process flows upward from individual supervisors, through the local safety office, and ultimately through the Agency level safety and health organizations. The process also provides for anonymous reporting of safety concerns that serves to by-pass all other reporting processes when there is fear of retribution for reporting more directly.

Question 10: Some of the most important risks seem to come from organizational factors and safety culture as some historical events indicate. What is done at NASA in this regard?

Answer 10 [response from Homayoon Dezfuli, NASA]: Please see the response to Q 2. Regarding our activities related to safety culture, NASA has an active Safety Culture Program. For more information, refer to http://sma.nasa.gov/sma-disciplines/safety-culture.

TECHNICAL SESSIONS
Tuesday, March 10, 2015, 3:30 p.m. – 5:00 p.m.

T7 Evaluating Residual Radioactivity in the Subsurface at Operating and Decommissioning Nuclear Power Plants

Session Chair: Jack Parrott, Senior Project Manager, Division of Decommissioning, Uranium Recovery and Waste Programs, NMSS/NRC, 301-415-6634, Jack.Parrott@nrc.gov

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

David Aird, Project Manager, Division of Risk Analysis, RES/NRC, 301-251-7926, David.Aird@nrc.gov

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

T8 Operational Radiation Protection and Environmental Monitoring at Nuclear Power Plants

Session Chair: Roger Pedersen, Senior Health Physicist, Division of Risk Assessment, NRR/NRC, 301-415-3162, Roger.Pedersen@nrc.gov

Session Coordinator: Micheal Smith, Health Physicist, Division of Risk Assessment, NRR/NRC, 301-415-3763, Micheal.Smith@nrc.gov

The questions below were not answered during the above session.

Question 1 [addressed to Michael Knochenhauer, Swedish Radiation Safety Authority]: Who manufactured the real time direct radiation monitoring system used in Sweden (i.e., hardware such as detectors and controllers)?

Answer 1 [response from Michael Knochenhauer, Swedish Radiation Safety Authority]: The contractor for the off-site gamma monitoring system is the company Scanmatic AS (www.scanmatic.no), a Norwegian company. Our formal counterpart was the Swedish subsidiary Scanmatic Environmental Technology AB (http://www.smetab.se).

T9 Status of the Level 3 PRA Project for Vogtle, Units 1 and 2

Session Chair: Kevin Coyne, Branch Chief, Division of Risk Analysis, RES/NRC, 301-251-7586, Kevin.Coyne@nrc.gov

Session Coordinator: Alan Kuritzky, Senior Reliability and Risk Engineer, Division of Risk Analysis, RES/NRC, 301-251-7587, Alan.Kuritzky@nrc.gov

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

T10 The Baseline Security and Force-on-Force Inspection Programs

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

Session Coordinator: Melissa Ralph, Technical Assistant, Division of Security Operations, NSIR/NRC, 301-287-3678, Melissa.Ralph@NRC.gov

The questions below were not answered during the above session.

Question 1: What confidence does the NRC have that the FBI will maintain the DTRA model in accordance with nuclear security safeguards handling protocol?

Answer 1: The NRC is confident the FBI ensures the law enforcement tactical team planning tools are protected in accordance with the standards set forth in Title 10 of the Code of Federal Regulations, Part 73, Sections 21-22. The NRC works closely with the FBI to confirm the tools are stored only in authorized locations and that they are accessible to only those law enforcement representatives who are trustworthy and reliable and have an official need to know the information.

Question 2: What lessons learned can you share about the Part 37 inspection activities that recently were halted?

Answer 2: There is a general need for clearer guidance on what is expected by licensees relying upon their existing security program under Part 73 to ensure protection of material covered by Part 37 (i.e., Category 1 and Category 2 radioactive material) as allowed under 10 CFR 37.11(b) of the regulations.

T11 Updated Spent Fuel Storage Renewal Guidance and Operating Experience

Session Chair: Aladar Csontos, Branch Chief, Division of Spent Fuel Management, NMSS/NRC, 301-287-9199, Aladar.Csontos@nrc.gov

Session Coordinator: Ricardo Torres, Materials Engineer, Division of Spent Fuel Management, NMSS/NRC, 301-287-0755, Ricardo.Torres@nrc.gov

The questions below were not answered during the above session.

Question 1 [addressed to Pamela B. Cowan, Exelon Generation]: Has Exelon done any cost-benefit analyses on the feasibility of a central dry cask storage site?

Answer 1 [answered to Pamela B. Cowan, Exelon Generation]: No.

Question 2 [addressed to Darrell Dunn, NRC]: It seemed some of the AMPs were focused on the used nuclear fuel. What expectations are there for inspecting the fuel and other components within the canister? Or will TLAAs be sufficient for in-canister components?

Answer 2 [response from NRC]: The interior cavity of dry storage canisters and bolted casks are not required to be inspected. The cavity and internals of canisters are carefully processed in preparation for dry storage, including moisture removal and a final backfill to establish an inert gas atmosphere. The requirements for loading operations mitigate degradation of the fuel, which was confirmed by lessons learned from a confirmatory long-term study on low-burnup fuel (see NUREG/CR-6745, “Dry Cask Storage Characterization Project-3 Phase 1; CASTOR V/21 Cask Opening and Examination” and NUREG/CR-6831, “Examination of Spent PWR Fuel Rods after 15 Years in Dry Storage”). These research results suggested that degradation of low burnup fuel cladding should not occur during the first renewal period. The US Department of Energy is sponsoring a similar confirmatory study on high-burnup fuel. The NRC requires licensees and CoC holders to have an aging management program (AMP) and uses license and CoC conditions to ensure the results from this program are evaluated in time to ensure safe storage of high burnup fuel. As part of the review of the renewal application, the NRC also evaluates any calculations or analyses, including time-limited aging analyses (TLAAs), if these provide the basis for conclusions related to the capability of the fuel assemblies to perform their intended safety functions. The staff makes a determination of whether these calculations and analyses support the requisite finding of reasonable assurance.

Question 3 [addressed to Darrell Dunn, NRC]: Would you talk about the “reactor” operational experience includes potential degradation of fuel cladding and bundles during reactor operation as well as canister operating experience?

Answer 3 [response from NRC]: The NRC considered the condition of the spent fuel and changes to the spent fuel cladding during irradiation in the reactor when developing requirements for storage. These changes are the reason for a distinction between high and low burnup fuel. The NRC-suggested conditions for storage are enumerated in Interim Staff Guidance (ISG)-11, Rev 3. This ISG specifies the maximum temperature that the fuel can experience and the need for a dry inert atmosphere in the storage container. This methodology mitigates degradation of the fuel cladding and other assembly components. The renewal application also assesses the effects of radiation embrittlement on steel components of the fuel assemblies that occur in the reactor as well as during storage. The NRC requires licensees and Certificate of Compliance (CoC) holders to conduct an aging management review of important to safety structures, systems and components for the renewal of specific licenses and CoCs. For aging effects that need an AMP, the NRC requires the licensee or CoC holder to evaluate past operational experience and identify how operational experience will be collected and utilized to inform the AMP in the renewal period. The NRC staff will continue to evaluate research and applicable operating experience to ensure loading and storage requirements remain adequate.

Question 4 [addressed to Kristopher Cummings, NEI]: For the license conditions associated with components internal to the canister (including the fuel), are there toll gates or AMPs for these components?

Answer 4 [response from NRC]: "Tollgates" have been included in renewal applications as an additional set of in-service assessments beyond the normal continual assessment of operating experience, research, monitoring, and inspections on dry storage system performance that is part of normal ISFSI operations for licensees during the initial storage period as well as the period of extended operation. An applicant may use tollgates to ensure the continued effectiveness of the Aging Management Programs (AMPs) for structures, systems and components, including the spent fuel assemblies. When issuing a license or CoC, the NRC considers whether conditions or technical specifications may be required to ensure critical elements of the AMPs, including specific criteria in tollgates, are effectively maintained. These conditions will be specific to information in the AMP described in the renewal application which staff relied upon to make the requisite safety findings of reasonable assurance of adequate protection of public health and safety, and the environment.

Question 5 [addressed to Kristopher Cummings, NEI]: Has there been a gaps analysis to ensure that the same level of public involvement is associated with ISFSI license renewal process as with the license amendment process?

Answer 5 [response from NRC]: The NRC ensures that the public has an opportunity to participate in NRC activities throughout our licensing processes, and Independent Spent Fuel Storage Installation (ISFSI) and Certificate of Compliance (CoC) renewals are no exception. Some ISFSI and CoC license renewals may involve public meetings where the public is provided an opportunity to ask Questions and provide comments. Furthermore, an opportunity for a hearing is generally offered upon an application for a specific ISFSI license renewal, just as with an application for an amendment to an ISFSI license. CoC renewals and CoC amendments are reviewed through a public rulemaking process. For CoC renewals, the NRC publishes a proposed rule in the Federal Register and invites the public to provide comments on the rule.

TECHNICAL SESSIONS
Wednesday, March 11, 2015, 10:30 a.m. – 12:00 p.m.

W12 Construction Inspection and ITAAC—How It All Comes Together

Session Chair: James Beardsley, Branch Chief, Division of Construction Inspection and Operational Programs, NRO/NRC, 301-415-5998, James.Beardsley@nrc.gov

Session Coordinator: Andrea Johnson, Reactor Operations Engineer, Division of Construction Inspection and Operational Programs, NRO/NRC; 301-415-2890, Andrea.Johnson@nrc.gov

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

W13 Operating Crew Performance during Extreme Scenarios: Lessons from Experiments and User Perspectives

Session Chair: David Desaulniers, Senior Technical Adviser, Division of Construction Inspection and Operational Programs, NRO/NRC, 301-415-5918, David.Desaulniers@nrc.gov

Session Coordinator: Jing Xing, Senior Human Performance Engineer, Division of Risk Analysis, RES/NRC, 301-251-7580, Jing.Xing@nrc.gov

The questions below were not answered during the above session.

Question 1 [addressed to John Stetkar, Stetkar & Associates]: Why wasn't Scenario 2 addressed by a Loss of Instrument Air sequence?

Answer 1 [response from John Stetkar, Stetkar & Associates]: I don't know. Since that event started with a Loss of Offsite Power, I assume that the operators initially focused on that condition. They quickly recognized the charging / letdown problem, but their procedures and training apparently did not completely address all of the effects from loss of instrument air. For example, during the post-event briefings, the operators noted that they thought that the first alternate letdown alignment was ineffective, because they did not see an increase in Volume Control Tank (VCT) level. The reason that VCT level did not increase was that an air-operated valve between the VCT and the alternate letdown connection to the charging pumps was in the closed position due to the loss of air. Letdown flow was being returned to the charging pumps, but VCT level could not increase because the VCT was isolated from that flowpath. The operators expected to see VCT level increase, and when it didn't, they concluded that the alternate letdown flowpath was blocked. If they were following a procedure for Loss of Instrument Air, it is apparent that the procedure did not alert them to the closed VCT valve or the fact that VCT level would not increase when they aligned the alternate letdown flowpath.

Question 2 [addressed to John Stetkar, Stetkar & Associates]: Operators are trained to respond on a symptom basis. Why are detailed narratives needed?

Answer 2 [response from John Stetkar, Stetkar & Associates]: Human reliability analysts account for the symptom-based Emergency Operating Procedures and the operators' training when they evaluate personnel performance during an event scenario. However, operating experience has shown that many factors can affect the operators' performance, such as stress, communications, failed equipment, availability and quality of Control Room displays, time limitations, possible conflicting priorities, etc. A detailed narrative of the evolving event scenario that describes the entire context of what is happening in the plant from the operators' perspective and all of the cues that the operators are receiving provides vital information for the human reliability analysts to perform their evaluations. These narratives are especially important for challenging scenarios that involve fires or flooding, partial losses of support systems, multiple equipment failures, and distractions like those during the two events that were summarized in the presentation.

Question 3 [addressed to John Stetkar, Stetkar & Associates]: Can you elaborate on the narratives? How do you define narratives? What are the important elements of narratives?

Answer 3 [response from John Stetkar, Stetkar & Associates]: Several references describe elements of a good narrative (although, in my personal opinion, few stress the importance of the narratives or provide good examples). In the simplest terms, the narratives are intended to describe what the operators are experiencing during the evolving event scenario. They establish the context for the human reliability analysis. They should describe what is happening in the plant and how the operators are alerted to the plant status. The timing of alarms, indications, and parameter trends are especially important, because they can affect what is foremost in the operators' attention at a given point in time. The narratives should not focus only on safety-related systems, "expected" operator performance according to the Emergency Operating Procedures, or the specific actions that are modeled in the PRA, because the operators' responses are based on the totality of their input, including expectations or needs to deal with problems that are not directly related to the PRA, but are important to stabilize the overall plant.

Question 4 [addressed to John Stetkar, Stetkar & Associates]: Given the complexities you've outlined in real-world situations, do you think contemporary HRA and PRA tools have any hope of accurately and meaningfully capturing these situations at the level of resources available for these analyses?

Answer 4 [response from John Stetkar, Stetkar & Associates]: Yes, I do. In PRAs, human reliability analysts, event sequence analysts, and plant operators should work together very closely when the models are being developed. By "very closely", I mean in direct real-time collaboration, not the serial process that has often been applied in the past. (In that process, event sequence analysts develop the models with some collaboration by operators and then describe the actions that the human reliability analysts must evaluate. The human reliability analysts then consult the operators about those actions, but often in isolation from the actual scenario context.) Several benchmark studies have shown that placing greater emphasis (and more resources) on initial development of comprehensive descriptions of the scenario context results in more realistic assessments of human performance and may reduce the amount of effort that is needed for those assessments. If the narratives are developed with close cooperation from experienced plant operators, they often do not require extensive effort. My own personal experience has also taught me that spending time to "tell the story" often forces me to think more clearly about nuances of the scenario that may affect personnel performance, but can be overlooked if I write only an abbreviated summary.

Question 5 [addressed to Andreas Bye, OECD Halden Reactor Project]: Knowledge based actions. Did you observe knowledge based actions? Is there a need to revise procedures to support knowledge based actions?

Answer 5 [response from Andreas Bye, OECD Halden Reactor Project]: The basic knowledge of the crew is important. We saw that in very difficult situations where there was a mismatch between the situation and the procedures, the crews went back to their basic knowledge in their decision making. So the Answer is yes, but also be aware of that the use of the procedures is closely linked to the conduct of operations and the definitions of the role of the procedures and the roles in the crew. If one knows the intents of the procedures and of the procedure steps, one may use the current procedures in a good way based on the conduct of operations.

Question 6 [addressed to Andreas Bye, OECD Halden Reactor Project]: Do all your crews have STA in the studies? Do all these crews have distinct STAs at their utilities at all times, or do they sometimes combine the STA and control room supervisor position?

Answer 6 [response from Andreas Bye, OECD Halden Reactor Project]: This was discussed thoroughly in the discussions in the session, based on the same Question posed orally. Filling in, I can say that we used the STA position actively in the last study, but not in the former ones. In the last study, all the crews had an STA in some of the scenario runs. The STA was in or out of the crew as part of the manipulation in the study. In the scenarios in which the STA was part of the crew, they were told to do as they would normally do at home. The STA role at the home plants vary.

Question 7 [addressed to Andreas Bye, OECD Halden Reactor Project]: Does the extent to which operators follow procedures depend on national cultural factors? Do operators from different countries behave differently?

Answer 7 [response from Andreas Bye, OECD Halden Reactor Project]: We do not compare different countries as such. There is quite different behavior between crews from different plants, and from different countries. It is important for us to know the conduct of operations for the plant that the crew is from, since these vary from plant to plant, and describe the roles in the crew. When it comes to variability in performance, we observed variability between crews from different plants and from the same plant, but also variability in the performance of one crew in different scenarios and situations. To get a little deeper in this matter, we did perform a study where we compared national and organizational cultures based on Hofstede's Questionnaires (not based on the performance in the scenarios). The objective was to look into generalizability of our studies rather than comparing nations as such. We compared U.S., Korean, and Swedish NPP operators. The results were that they were more similar than expected based on the Hofstede predicted profiles. This could be early evidence of a common industrial culture, which is promising with respect to the cross-cultural generalizability of simulator research. Results on this can be found in Halden Work Report, HWR-1027.

Question 8 [addressed to Andreas Bye, OECD Halden Reactor Project]: Have you done similar studies with BWR simulators? If so, do you see any significant differences in EOP approaches?

Answer 8 [response from Andreas Bye, OECD Halden Reactor Project]: We did a similar study in 2004 with a BWR simulator. BWR EOPs are different from PWR, but we saw similar results in that study: When complexity in the form of time pressure, information load and masking were higher, the performance of the crews degraded. The results are documented in Halden Work Report, HWR-757.

Question 9 [addressed to Andreas Bye, OECD Halden Reactor Project]: Was there any significant variances in the results with respect to procedural compliance when comparing BWR and PWR operating crews?

Answer 9 [response from Andreas Bye, OECD Halden Reactor Project]: We have not compared BWR and PWR operating crews in this respect. For clarification, it has never been any Question of *not* following procedures in any of our studies as we have seen. All crews from all nuclear power plants follow procedures. What we have studied is situations in which the procedures do not fit the situation, and then it is more a Question of different crew strategies for how to get to the right procedure, how to know whether you are in the right procedure, and how to get out of one procedure and to the right procedure in the case that you know that you're in the wrong one. In this case it was especially interesting to study the effect of better access to procedure backgrounds and intents.

Question 10 [addressed to Andreas Bye, OECD Halden Reactor Project]: Were scenarios run on advanced control room with computer-based displays? Do they help? Is there a big difference in performance between standard and computer based displays?

Answer 10 [response from Andreas Bye, OECD Halden Reactor Project]: All scenarios were run in HAMMLAB (HAlden Man-Machine LABoratory) with computer-based displays. Our "standard" set of displays on the PWR simulator is made to be as functionally similar to the home plant control room (which is panel based) as possible. In some cases we test new display solutions. E.g., in the last study we used an advanced overview display as well as a procedure background tool. The data is still under analysis, but a first impression is that they do help. Whether there is a big difference in performance between standard and computer based displays depends on each solution. It is fully possible to design bad computer-based displays.

Question 11 [addressed to Andreas Bye, OECD Halden Reactor Project]: Did the experiments provide data, and was it analyzed about the challenges in communication among crew members using digital interface? "Keyhole effect"

Answer 11 [response from Andreas Bye, OECD Halden Reactor Project]: The keyhole effect is normally used about an effect of early digital interfaces where one had few displays and had to dive into several layers of interfaces thereby losing the overview. Also, this can harm the communication between the crew members as indicated, since each operator may dive into details and the other crew members don't know what the others are doing. However, this depends on each solution, and is not an effect that will occur for all digital interfaces. For example, in most of our configurations in the control room we use a large screen overview display that is designed in order to create a common overview of the crew. In the last experiment some of the scenario runs did not have the overview display in use, and there may be results in this direction there. These results are under analysis and will be published later.

Question 12 [addressed to Andreas Bye, OECD Halden Reactor Project]: Based on use of EOPs, and the shortcomings observed, is it better to from a human performance perspective to go to functional recovery procedures which address symptoms or safety function challenges instead of event based EOPs?

Answer 12 [response from Andreas Bye, OECD Halden Reactor Project]: We have not compared such different kinds of procedures as such, but we are using the basic procedure set that is used at the home plant of the simulator. Most plants have gone in the direction that you imply, for example the Westinghouse procedures are a combination of symptom-based and event-based procedures.

Question 13 [addressed to Andreas Bye, OECD Halden Reactor Project]: Would you discuss if any research activities of your project regarding an in-situ simulation/analysis computer program or tool that can help operators identify actual event scenarios occurring?

Answer 13 [response from Andreas Bye, OECD Halden Reactor Project]: We have been studying some sorts of "what-if" simulation tools. In one case an automatic procedure execution tool that intended to give the operators an idea of in what direction the plant would go in following a procedure, another case was a simulation of accidents. These were not directly for operators to identify events though.

Question 14 [addressed to Andreas Bye, OECD Halden Reactor Project]: Any comment regarding use of flowchart path versus EOP procedures? HB Robinson used the path – very useful for look-aheads.

Answer 14 [response from Andreas Bye, OECD Halden Reactor Project]: We have not compared different procedure types directly in an experiment. However the flowchart tool we made for procedure backgrounds seemed good.

Question 15 [addressed to Ron Gibbs, STPNOC]: Do the lessons learned from the way the STA performed when away from the operators suggest that control rooms have to be redesigned?

Answer 15 [response from Ron Gibbs, STPNOC]: No. That part of the study (STA isolated from crew) was intended to and did show that when the STA is acting independently, he is effective in analyzing the data, event sequence and recommending appropriate actions/ strategy/ priorities (more tactical). When the STAs were allowed back in the control room, the STA become involved in step-by-step tasks and losing his independence/tactical view. The degree of involvement varied by crew. We want the STA to be able to provide input, so we do want him close. What this study showed me is that we have to re-evaluate our expectations for our STAs and other changes to help maintain their role. Our current event response model has the STA in the middle of the control room (due to location of procedures) making him too accessible and more interactive with the crews. This keeps him from being tactical and evaluating overall mitigation strategies, priorities, etc. What we are looking at is physically moving the STA out of the middle of the control room and giving him the necessary tools so that he can be more tactical in analyzing the event - looking forward, looking for & resolving conflicting data (indications or procedure data).

Question 16 [addressed to Ron Gibbs, STPNOC]: Much of what you describe seem to fit in the “Operator Fundamentals” bucket. What are you doing to improve operator fundamentals? Are you just relying on deeper CPE style simulator scenarios?

Answer 16 [response from Ron Gibbs, STPNOC]: We focus on operator fundamentals constantly. All of our events are broken down to the fundamental level to show where improvement is needed. What our participation in the Halden Project has shown us is that we may have focused on the wrong fundamental(s) had we not participated. In 2011, we took away that we needed focus on EOP bases/ background documents (fundamental - knowledge). In hindsight, this was probably only part of the issue - defined roles & responsibilities (fundamental - teamwork) the other part.

This study helped us identify which fundamental needed improvement. We have implemented more of the CPE style scenarios for the many aspects that they bring. Knowledge - going deeper into the EOPs to increase knowledge of bases. Teamwork - roles & responsibilities. As discussed above, we will be monitoring the STA for their performance. Also, this allows the Shift Manager to monitor his crew performance. When he sees a decline in performance, the chance to intervene to correct. We have given the SM one more tool (short break) to improve crew performance.

Question 17 [addressed to Ron Gibbs, STPNOC]: The use of rest periods for operators. Are you looking into the guidelines used by air traffic controllers?

Answer 17 [response from Ron Gibbs, STPNOC]: I apologize for the confusion. We are under the NRC mandated "Fatigue Rule" which is very similar to the air traffic controllers. When we used the term, 'fatigue' here we were referring to a cognitive/ mental fatigue caused by the high mental workload during the initial stages of an event. We are trying short breaks to 'reset' our operators mentally as a way to compensate for this.

Question 18 [addressed to Ron Gibbs, STPNOC]: Have your insights gained from these studies altered how you initially train your new licensed operators?

Answer 18 [response from Ron Gibbs, STPNOC]: Not at this time. We are considering introducing some of the potential technology advances (EOP usage tool) into the LOT arena.

We believe that this would improve initial understanding of bases documents & mitigation strategies.

Question 19 [addressed to Ron Gibbs, STPNOC]: Do you think the added emphasis on EOPs is (could) create an unintended consequence of not understanding/ being able to handle more frequent, less severe events?

Answer 19 [response from Ron Gibbs, STPNOC]: No, due to the operator program being an accredited program, the frequent less severe accidents are required to be trained on the same periodic frequency. We have not diverted any requal time from the required training.

Question 20 [addressed to Jing Xing, NRC]: Who are you partnering with for the collection of data for SACADA? What benefit do they get? Can other utilities join the effort?

Answer 20 [response from Jing Xing, NRC]: NRC currently partners with the South Texas Project Nuclear Operating Company (STPNOC), the Halden Reactor Project (HRP), Norway, and the Korea Atomic Energy Research Institute (KAERI) on using SACADA to collect licensed operator simulator training and experiment data. The NRC is discussing collaboration with The Advanced Test Reactor (ATR) at the Idaho National Laboratory, Taiwan Power Company (TPC), and the Nuclear Research Institute (UJV), Czech Republic are expected to partner with NRC soon. The NRC welcomes domestic and international utilities to partner with the NRC on SACADA. Contact Y. James Chang (301-251-7589; James.Chang@nrc.gov) for more information.

Question 21 [addressed to Jing Xing, NRC]: Would you share your view on potential implications of aging-related degradation of components and systems to human reliability analysis, including validation of procedure adequacy?

Answer 21 [response from Jing Xing, NRC]: Aging-related degradation of components and systems could potentially impact human reliability in assessing plant status because the perceived information about the plant systems may or may not accurately and stably represent the actual information. HRA methods do not have explicit guidance on the consideration of aging-related I&C degradation. The new NRC HRA method, IDHEAS, has one crew failure mode “Critical data misleading or not available,” one of the major causal factor for this failure mode is “Unreliable source information;” In evaluating this failure mode, analysts should examine factors that may lead to information unreliable, such as degraded I&C.

None of the existing HRA methods explicitly guide analysts to consider the implication of degraded I&C on the validation of procedure adequacy. In most HRA, the analysis of procedure adequacy focuses on assessing whether the procedures is detailed enough for the scenario, without considering the procedure assumption that the source information is reliable and stable.

The NRC research staff developing the IDHEAS method will consider these issues as part of the development efforts.

W14 Optimizing Waste Disposal for the New Millennium

Session Chair: Andrew Persinko, Deputy Director, Division of Decommissioning, Uranium Recovery, and Waste Programs, NMSS/NRC, 301-415-7479, Andrew.Persinko@nrc.gov

Session Coordinator: Gregory Suber, Branch Chief, Division of Decommissioning, Uranium Recovery, and Waste Programs, NMSS/NRC, 301-415-8087, Gregory.Suber@nrc.gov

The questions below were not answered during the above session.

Question 1 [addressed to Christianne Ridge, NRC]: Is there any equivalent IAEA guidance to the BTP? If not, how do other countries perform similar averaging activities and do they have an intruder scenario?

Answer 1 [response from Christianne Ridge, NRC]: IAEA has standards (SSR-5) and guidance (SSG-23) for intruder protection and specifically recommends that hot spots in waste be assessed to ensure continued protection of an inadvertent intruder. The CA BTP is consistent with the IAEA guidance. Information on how other countries address concentration averaging is limited. Many countries store sealed sources, but a few dispose of them as well. We have found few specifics on such disposals. The National Reports for the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management are silent on most hot spots other than sealed source, and often have limited information on sealed sources.

The most extensive hot spot guidance from another country that the NRC staff is aware of is the guidance for the Low Level Waste Repository, near Drigg, Cumbria. Because this site expected to undergo coastal erosion in the absence of intervention, the regulator considered specific intrusion scenarios related to an individual picking up a radioactive discrete item from the beach below the site. Guidance for the site from the UK Environment Agency states that an optimized approach is likely to entail limiting the radioactivity of discrete items disposed of at the site, and preventing any processes that might lead to the production of high dose particles.

Question 2 [addressed to Christianne Ridge, NRC]: Should we attach any significance to the fact that the Branch Technical Position (BTP) is just that, a “BTP” and not a NUREG/RG? (i.e., are the staff’s positions in the BTP not final?)

Answer 2 [response from Christianne Ridge, NRC]: The CA BTP was published as a Branch Technical Position because it is a revision of a document that has historically been a Branch Technical Position. There is no significance to its publication as a BTP rather than a NUREG or Regulatory Guide. The staff’s positions are final, although as stated in the *Federal Register* Notice publicizing the release of the BTP, the staff recognizes that in the future, issues could arise that may need to be clarified in a generic publication such as a Regulatory Issue Summary (RIS).

Question 3 [addressed to Lisa Edwards, EPRI]: The EPRI study was based on Pacific Northwest National Laboratory data found in NUREG-6537(?) which was published about 20 years ago and only included ~40 samples. Are you comfortable using this data and if so, why?

Answer 3 [response from Lisa Edwards, EPRI]: Of course it would be great to have more and more recent samples, but this is a fairly comprehensive data set that is largest set I am aware of. It covers both PWRs and BWRs and includes varying degrees of fuel integrity. It is also no older than the data sets used to develop other codes (such as the gale code). We are comfortable that the data from these samples is representative of the industry waste streams particularly when production mechanisms are considered. This will be covered in more detail in the written report. New samples are very expensive (in excess of $50K) to have analyzed as well, but if new samples become available, EPRI would consider reviewing those results and including them in an updated analysis.

Question 4 [addressed to Rod Baltzer, Waste Control Specialists, LLC]: How does SAFSTOR of shut down plants affect volume of waste expectations at LLW sites?

Answer 4 [response from Rod Baltzer, Waste Control Specialists, LLC]: It won't impact the volume as much as the timing of the volumes.

Question 5 [addressed to Rod Baltzer, Waste Control Specialists, LLC]: EnergySolutions is considering an option to reopen Barnwell. If they do, how will that impact WCS' plans?

Answer 5 [response from Rod Baltzer, Waste Control Specialists, LLC]: Given the recent news conference held by the Governor of South Carolina, it is highly doubtful that Barnwell will reopen. WCS believes that our facility is better designed for Class B/C disposal than Barnwell and would be a better option for generators trying to dispose of their LLW and reduce their potential future liability.

Question 6 [addressed to Rod Baltzer, Waste Control Specialists, LLC]: This might be premature, but what kind of packaging requirements do you anticipate for GTCC disposal?

Answer 6 [response from Rod Baltzer, Waste Control Specialists, LLC]: It probably is a little premature, but we expect to handle GTCC similar to our current Class B/C disposal. You can see our waste acceptance criteria for packaging on our website at www.wcstexas.com.

Question 7 [addressed to Rod Baltzer, Waste Control Specialists, LLC]: Can WCS take foreign waste at the exempt cell?

Answer 7 [response from Rod Baltzer, Waste Control Specialists, LLC]: Yes, we can take foreign waste at our exempt (hazardous) waste landfill.

Question 8 [addressed to all]: Will a GTCC facility in Texas need to be licensed by the NRC?

Answer 8 [response from Melanie Wong, NRC]: We have received a request from the Texas Commission on Environmental Quality (TCEQ) asking a similar Question. We are still evaluating TCEQ's request at this time. Our response will be made publicly available once our evaluation is complete.

W15 Regional Session

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

Session Coordinator: Steve Barr, Senior Emergency Preparedness Inspector, Division of Reactor Safety, RI/NRC, 610-337-5316, Steve.Barr@nrc.gov

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

W16 The Future of Risk-Informed Regulation

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

Session Coordinator: CJ Fong, Reliability and Risk Analyst, Division of Risk Assessment, NRR/NRC, 301-415-8474, cj.fong@nrc.gov

The questions below were not answered during the above session.

Question 1: How does NRC assess Safety Culture and input its PRA calculations, especially in light of the recent NRC policy statement on Safety Culture?

Answer 1 [response from Joseph Giitter, NRC]: NRC does not assess safety culture with the use of an input derived from Safety Culture assessment to PRA calculations. Rather, NRC uses the risk-informed Reactor Oversight Process which relies significantly on PRA calculations to determine how the NRC inspectors should engage in assessing licensee's Safety Culture. This approach implicitly recognizes the important relationship between Safety Culture and Nuclear Safety. For example, in the event ROP assessments result in a particular plant entering Column 4 (Multiple\Repetitive Degraded Cornerstone), per Inspection Manual Chapter 95003, NRC may request that licensee to conduct a Safety Culture assessment using a third party independent assessor and expend significant inspection efforts on a graded safety culture assessment based on the results of that third party assessment. The Safety Culture Policy

Statement is not a regulation. It is the Commissions' expectation regarding licensees' safety culture.

Question 2: Why shouldn't NRC allow plants to use qualified PRA models for the risk-prioritization initiative (RPI) of cumulative effects to prioritize and in some cases remove projects required by regulations that do not make plants safer?

Answer 2 [response from Joseph Giitter, NRC]: The draft Commission paper provides several options. The option that staff has recommended for the Risk Prioritization Initiative would allow licensees to use their current and available probabilistic risk assessment information to prioritize regulatory issues and to request risk-informed schedule changes. The process as envisioned does not allow for the elimination of issues. Staff does not believe that we have enough experiences and the necessary regulatory structure to enable licensees to use their models to remove projects. However, licensees can use other existing well-established processes to request exemptions or license amendment changes if they feel the issue is of low or very low safety significance.

Question 3: Why not strengthen PRA applications by using the risk prioritization initiative of projects (part of cumulative effects)?

Answer 3 [response from Hossein Hamzehee, NRC]: The Risk Prioritization Initiative (RPI) is focused on allowing licensees to use plant-specific information to prioritize regulatory and plant-initiated activities thus focusing time, attention, and resources on issues of the highest safety significance. In addition, RPI has been observed in the demonstration pilots to further the use of probabilistic risk assessment (PRA). Therefore, in a way the initiative incentives the further use of PRA and can support additional benefits of using PRA.

Question 4: Two barriers to increased use of RI regulation:

1. The deterministic "rules" have the backing of federal law while "risk" thinking has the backing of "policy." They therefore are not on an equal regulatory basis, which impacts their "balance."
2. Many subject matter experts in the NRC grew up in the deterministic world are intragent [sic] …unwilling or unable to think beyond the deterministic view.

What can be done to [illegible]…to tap into their expertise yet get them to think more broadly? Kuhn's theory of scientific revolutions would suggest we need to wait until they die off – that is not acceptable.

Answer 4 [response from Hossein Hamzehee, NRC]: NRC's approach to risk-informed regulation uses risk insights together with other factors to make regulatory decisions, with the goal of providing reasonable assurance of adequate protection of public health and safety without undue burden on licensees. The "other factors" include traditional engineering analyses and concepts such as defense-in-depth, safety margin, the single failure criteria, and fault-tolerant/fail-safe designs. The risk-informed approach also includes monitoring of advances in science and engineering and of operating experience/performance so that new issues can be identified and addressed before an undesirable outcome is reached.

NRC has made great strides in moving towards risk-informed regulation for nuclear power reactors. A number of key regulations have been written that incorporated risk insights, and several alternative regulations that reduce licensee burden are the result of risk insights. The NRC staff is well aware of the Commission's policy regarding the use of risk insights to the extent warranted by the issue being considered and the state-of-the-art in risk assessment technology.

It may appear that NRC's progress towards risk-informed regulation is slower than necessary. One reason for this perception might be a focus on the risk numbers and not on the "other factors" mentioned above. It is the focus on these other factors that distinguishes NRC's risk-informed approach from one that is risk-based. It is also true that many regulations are not amenable to a risk-informed treatment, either because of limitations in current risk assessment methods or because the subject of the regulation itself is better addressed in another manner. Regulations regarding financial obligation, fitness for duty, and minimum staffing requirements are examples of regulations that are not likely to be risk-informed for these reasons.

While Thomas Kuhn opines that scientific revolutions occur when an anomaly "subverts the existing tradition of scientific practice," the NRC staff does not believe that the transition from "deterministic" to "risk-informed" rises to the level of the major scientific paradigm shifts studied by Dr. Kuhn. On the contrary, the blending of risk-insights with traditional engineering has been taking place in an evolutionary manner at the NRC for many years, and will continue to take place in the future.

Question 5: There is compelling evidence that LERF is a poor metric to approximate early health risk. The ad hoc definition is vague. Perhaps, like the Europeans, LRF is a better surrogate.

Fukushima did not experience an early release so LERF = 0 for that event. If might even be possible that the releases of Fukushima did not reach the "limits" for latent health effects. We can all agree that Fukushima did not reflect an acceptable event. My Question: is NRC actively working to modernize the surrogate metrics used to reflect "adequate protection" of the public? Candidates could include land contamination metrics along with [obtainable?] plant performance metrics. See for example Douglas Hibbard's caution w/ the failure of risk management that effective risk management can only be made via independently [observable?] metrics.

Answer 5 [response from Hossein Hamzehee, NRC]: The NRC notes that LERF is a surrogate for the Commission's quantitative health objective for early fatalities. The definition of LERF may be found in SRM-SECY-98-0144, "White Paper on Risk-Informed and Performance-Based Regulation:"

> LERF is the frequency of those accidents leading to significant, unmitigated releases from containment in a time-frame prior to effective evacuation of the close-in population such that there is a potential for early health effects.

Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," defines LERF as follows:

> [LERF] ... is defined as the sum of the frequencies of those accidents leading to rapid, unmitigated release of airborne fission products from the containment to the environment occurring before the effective implementation of offsite emergency response and protective actions such that there is the potential for early health effects. Such accidents generally include unscrubbed releases associated with early containment failure shortly after vessel breach, containment bypass events, and loss of containment isolation.

The relationship of LERF to the Quantitative Health Objectives (QHOs) is also stated in RG 1.174: "In this context, LERF is being used as a surrogate for the early fatality QHO. This definition [of LERF] is consistent with accident analyses used in the safety goal screening criteria discussed in the Commission's regulatory analysis guidelines."

While the Fukushima event is not considered an "acceptable" event, it should be noted that there have been no prompt fatalities resulting from radiation resulting from that event. In this sense, Fukushima does not contradict the relationship between LERF and the prompt fatality QHO.

It should be pointed out that current NRC guidance for performing regulatory analyses includes consideration of onsite and offsite property damage. The NRC did again consider economic consequences, including land contamination, following the Fukushima event. In SRM-SECY-12-0110, "Consideration of Economic Consequences within the U.S. Nuclear Regulatory Commission's Regulatory Framework," the Commission found that "... economic consequences should not be treated as equivalent in regulatory character to matters of adequate protection of public health and safety." The Commission approved the NRC staff's recommendation to enhance the currency and consistency of the existing framework through updates to guidance documents integral to performing cost-benefit analyses in support of regulatory, backfit, and environmental analysis.

The NRC's use of core damage frequency and LERF as quantitative surrogates for the QHOs would seem to agree well with the Questioner's reference to Douglas W. Hubbard, (presumably from his book, "The Failure of Risk Management: Why It's Broken and How to Fix It"), because these surrogates are objective and are readily quantified with risk tools currently in use.

In summary, LERF is an objective surrogate for the prompt fatality QHO, the definition above is well-understood by risk practitioners developing nuclear power plant risk models, and there is no new information or operating experience that would indicate a change is necessary in this area.

Question 6: Regarding the Vogtle 4b LAR, the licensee submitted a “seismic penalty” approach vs. SPRA to quantify worst case seismic impact to RICT and reduce RICT limit by the seismic penalty value. Does the NRC endorse this approach for future 4B LARs?

Answer 6 [response from Hossein Hamzehee, NRC]: Since the Vogtle 4b LAR is still under staff‘s review, it is more appropriate to address this Question after the staff issues the license amendment and the SE is publically available.

Question 7: Can you clarify the safety benefit of a relocation of surveillance frequency requirements out of TS into a licensee controlled program? How do you evaluate this in PRA?

Answer 7 [response from Hossein Hamzehee, NRC]: The potential safety benefit of relocating TS surveillance frequencies is that an optimum interval can be identified that limits unnecessary testing, and therefore limits unavailability and unnecessary wear and tear on SSCs. The unavailability of an SSC is typically modeled as a basic event in PRA and should reflect actual plant testing and maintenance practices. If these testing and maintenance practices (e.g., TS surveillances) are changed, the PRA model will reflect that change.

Question 8: Is there any discussion for LARs where the issue is seismic but a seismic PRA does not exist? But using risk techniques the increase in delta CDF is negligible can be shown?

Answer 8 [response from Hossein Hamzehee, NRC]: The guidance on risk-informed LARs (RG 1.174) states that while all modes and hazard groups (e.g., seismic) must be *addressed*, it is not always necessary to have a PRA of such scope. Instead, qualitative arguments or bounding analysis (e.g., SMAs) may be used provided that adequate technical justification is provided.

Question 9: Can the NRC elaborate on any research or collaboration that may be on-going relating the FPRA assumption and modeling and how research findings will be incorporated into NRC guidance documents? Timing?

Answer 9 [response from Hossein Hamzehee, NRC]: The NRC has many research activities and collaborations ongoing through its Office of Nuclear Regulatory Research (RES). Among these activities are research in Very Early Warning Fire Detection Systems (VEWFDS), heat release rates from fires, and fire frequency and suppression values. The standard procedure is for RES to produce a NUREG report to document this research.

NRR generally approves guidance for regulatory use in Regulatory Guides. The timing associated with these Guides depends on when the research product is issued, and the need for the product to support NRR’s regulatory decision.

Question 10: Are you also considering risk-informed regulations for digital I&C? If so what is the approach or strategy?

Answer 10 [response from Hossein Hamzehee, NRC]: Digital Instrumentation and Control Probabilistic Risk Assessment - The objective of this research is to identify and develop methods, analytical tools, and regulatory guidance for (1) including models of digital systems in nuclear power plant PRAs, and (2) incorporating digital systems in the NRC's risk-informed licensing and oversight activities.

Research Approach - The NRC has been investigating reliability modeling of digital systems, which encompasses both hardware and software, for several years. Previous projects identified a set of desirable characteristics for reliability models of digital systems and assessed candidate methods against these attributes. In the area of digital hardware reliability, a simulation-based tool has been developed to determine the combinations and sequencing of component level failures that could impact system functions. Current research efforts are focused on developing methods for quantifying software reliability. As an initial step in this area, an expert panel was convened to establish a philosophical basis for modeling software failures in a reliability model. After reviewing several quantitative software reliability methods, two methods to apply to an example software-based protection system in a proof-of-concept study: the Bayesian Belief Network (BBN) approach and the statistical testing method. These methods are being applied to the Loop Operating Control System (LOCS) of the Idaho National Laboratory (INL) Advanced Testing Reactor (ATR). The work has highlighted several areas for additional research for PRA modeling of digital systems is needed, including the following:

- Defining and identifying failure modes of digital systems and determining the effects of their combinations on the system.
- Methods and parameter data for modeling self-diagnostics, reconfiguration, and surveillance, including using other components to detect failures.
- Data on hardware failures of digital components, including addressing the potential issue of double-crediting fault-tolerant features, such as self-diagnostics.
- Data and methods for modeling common-cause failures (CCFs) of digital components.
- Methods for addressing human reliability and modeling uncertainties in modeling digital systems.

Even if an acceptable method is established for modeling digital systems in a PRA and progress is made in the above areas, (1) the level of effort and expertise required to develop and quantify the models will need to be practical for vendors and licensees, and (2) the level of uncertainty associated with the quantitative results will need to be sufficiently constrained so that the results are useful for regulatory applications. Therefore, a goal of this research program is to assess the practicality and usefulness of including digital systems in nuclear plant PRAs.

Recent accomplishments and near term objectives include the following:

- Completed development of a failure mode taxonomy for digital I&C system performed by the OECD/NEA Working Group on Risk Assessment (WGRISK) (NEA/CSNI/R(2014)16, "Failure Modes Taxonomy for Reliability Assessment of Digital I&C Systems for PRA".)

- In collaboration with the Korea Atomic Energy Research Institute, work is ongoing to quantify software reliability using BBN-based on software development cycle quality attributes.
- Ongoing work to estimate the reliability, including software, of the ATR LOCS using PRA-based statistical testing.

For More Information Contact Ming Ling, RES/DRA, at Ming.Ling@nrc.gov

Question 11: 1) Regarding seismic PRAs to be submitted for NTTF Re 2.1, what are the NRC's plans to form an intra-agency review between NRO, NRR, RES of the suture SPRAs against the NRC-endorsed EPRI SPID document and RG 1.200? 2) Which NRC department will have lead review role?

Answer 11 [response from Hossein Hamzehee, NRC]: The first set of licensee submittals to provide the results of a seismic risk evaluation (seismic probabilistic risk assessment, SPRA, or seismic margins analysis, SMA) is scheduled for mid-calendar year 2017, with another set in 2019 and the final one in 2020. The actual approach to the NRC staff's review of these submittals will likely be refined several times before they start to arrive.

However, it is likely that a "matrix approach" will be used, similar to what is currently being done for the NRC staff reviews of the Expedited Seismic Evaluation Process (ESEP) submittals. For the ESEP submittal review, NRC created a matrix organization that includes staff members from the Office of Nuclear Regulatory Research, the Office of New Reactors (NRO), the Office of Nuclear Reactor Regulation, and contractors. The management oversight for the ESEP review is provided by a Branch Chief within NRO and the technical oversight by a panel of senior NRC staff members from the various Offices and contractor individuals.

While the approach for the NRC staff review of the SPRA or SMA submittals may change between now and when the submittals are received, NRC management will take steps to ensure that individuals with the necessary qualifications and experience are assigned to perform the reviews, irrespective of the Office in which the individual works.

Question 12: In your example you mentioned some SSC[s] being classified as safety significant that in fact are not and the opposite [is true]. This biases the deterministic analysis. But why not modifying the safety classification of SSCs? [sic] At least for SSCs which are not safety classified and that should be. [sic]

Answer 12 [response from Hossein Hamzehee, NRC]: In a PRA, the risk significance of nuclear power plant structures, systems, and components (SSCs) is determined, not the safety significance. Using the process outlined in 50.69, licensees can then use the risk-significance, along with deterministic insights, to determine the safety class of structures, systems, and components (SSCs). The four risk-informed safety classes (1-4) are used to determine the applicable QA requirements. For more information, see: http://www.nrc.gov/reading-rm/doc-collections/cfr/part050/part050-0069.html

Question 13: There is a potential disconnect between "reducing conservatisms" and use of "publically available data" when licensees incorporate proprietary information into LARs. How do you balance these principles?

Answer 13 [response from Hossein Hamzehee, NRC]: Regarding reducing conservatism vs. use of proprietary information that may not be publicly available, recognize that licensees are free to include any information that they choose in their LRAs/LARs, but have the right to declare whatever information they choose to be proprietary. NRC would publish the results of the analysis that uses such information, but would not make the information itself publicly available. Therefore, licensees can reduce conservatism via proprietary information without the concern that it could compromise safety/security by becoming publicly available.

Question 14: Do you think that RoverD and the simplified approach are licensable as design basis as they are not true PRA? Discuss the difference between PRA and risk-informed.

Answer 14 [response from Hossein Hamzehee, NRC]: The risk-over-deterministic ("RoverD") approach to evaluating the risk impact of GSI-191 is currently under review as part of the South Texas Project (STP) pilot. A similar methodology, sometimes referred to as "the simplified approach" was described to the staff by industry representatives at a public meeting on July 22, 2014.

The review of the STP pilot is on-going and no licensee has submitted a license amendment request referencing the simplified approach. Therefore, the staff has not received the detailed technical information needed to assess the acceptability of the methods.

Regarding the difference between the terms "PRA" and "risk-informed", various formal definitions exist for both terms (See NUREG-2122). In general, a PRA is a systematic method for assessing the likelihood of accidents and their potential consequences. For something to be risk-informed, it must combine risk information (which may be derived from a PRA) with other factors such as traditional engineering insights.

Question 15: What progress do you foresee in the application of RI methods to addressing external events as part of [the] Fukushima effort?

Answer 15 [response from Hossein Hamzehee, NRC]: Even without the Fukushima event, risk assessment technology has been moving forward. The Commission's PRA Policy Statement sets forth the Commission's expectations that PRA be used in regulatory decision making to the extent supported by the state of the art. The Commission's "phased approach to PRA quality" says that risk-informed decisions should be based on risk assessments where the aspects important to the decision follow NRC endorsed consensus standards on PRA. The latest endorsed PRA standard includes external hazards, so that the NRC anticipates increased use of PRA models to evaluate the risk of external events.

As a result of the Fukushima event, licensees were required to re-evaluate their design basis protections against seismic and external flooding hazards (NTTF Rec. 2.3). In addition, the

NRC requested licensees to further evaluate seismic and flooding hazards to the latest methods used for licensing new reactors (i.e., beyond the current design basis for the plant) (NTTF Rec. 2.1). The seismic evaluations used probabilistic seismic hazard methods to determine whether a licensee should perform a seismic risk evaluation for its site. This will result in a number of plants having seismic PRA models that meet the current PRA standard. While the flooding evaluation methods are also the latest methods, they tend to be deterministic rather than risk-informed, because additional work is needed to develop risk assessment methods for external flooding. This has led to an increased focus in the risk community on developing such methods, including probabilistic flooding hazard approaches and calculation of structure, system and component fragility to flooding events.

In summary, NRC expects increased application of risk-informed methods in addressing external events, not only because of Fukushima but because it is Commission policy.

Question 16: Oconee's 805 pilot was approved only after it withdrew its FPRA and pledged completion of a major modification within a few years. It later failed to complete the modification but was allowed to retain its 805 status (without "self-approval") still pending completion of the modification (now several years later). How does NRC justify allowing Oconee to retain this status vs. the original appendix R basis? How are similar situations being addressed for "non pilots?"

Answer 16 [response from Hossein Hamzehee, NRC]: By letter dated July 1, 2013, the NRC issued a Severity Level (SV) III violation and confirmatory order to Duke for not meeting the license condition for completing the Protected Service Water (PSW) system. Duke has provided incremental milestones (ML13079A321) for the completion of the PSW system and listed the compensatory measures in place to manage fire risk. At the time of the violation, no civil penalties were assessed (primarily because of Duke's corrective actions and the NRC's use of a confirmatory order); however, civil penalties will be considered as part of enforcement if Duke does not meet these incremental milestones.

In addition, while completing transitioning their fire protection program to NFPA 805, Duke continues to provide adequate protection of the public, due to the compensatory measures in place at Oconee. Although the final completion date for PSW is 2016, the NRC issued the confirmatory order with intermediate milestones to ensure incremental progress was made. These intermediate milestones provided significant risk reduction prior to 2016.

Question 17: Will the US fleet ever develop level 3 PRAs on a widespread basis? If so, when? 5,10, 15 years?

Answer 17 [response from Hossein Hamzehee, NRC]: For plants operated with licensees under 10 Code of Federal Regulations (CFR) Part 50 (this includes all currently operating plants) or future plants licensed under 10 CFR 50 (Watts Bar Unit 2 is the only plant currently in this category) there are no regulatory requirements to have and maintain a Level 1, 2, or 3 PRA.

For plants licensed under 10 CFR Part 52 there is a requirement to have Level 1 and Level 2 PRAs. This requirement is found in 10 CFR Parts 50 and 52. Part 50.71(h)(1), states: "No later than the scheduled date for initial loading of fuel, each holder of a combined license under subpart C of 10 CFR part 52 shall develop a level 1 and a level 2 probabilistic risk assessment (PRA). The PRA must cover those initiating events and modes for which NRC-endorsed consensus standards on PRA exist one year prior to the scheduled date for initial loading of fuel." It should be noted that NRC-endorsed consensus standard exists for Level 1 and the Level 2 standard has been issued for trial use. The standard Level 3 standard is nearing completion. Therefore, for Part 52 plants also there is no requirement for a Level 3 PRA.

Even though there are no regulatory requirements to develop Level 3 PRAs, NRC is interested in the potential insights that can be gained from Level 3 PRAs. Therefore, the NRC's Office of Nuclear Regulatory Research (RES) is currently developing a detailed all modes and all hazards Levels 1, 2 and 3 PRA based on Vogtle Units 1 and 2. Insights from this project may lead to changes in the PRA requirements and motivate some licensees to develop Level 3 PRAs.

In summary, the NRC staff recognizes Level 3 PRA as an important tool, and yet, the NRC staff unable to predict when the US Fleet will develop/deploy Level 3 PRAs on a wide spread basis.

Question 18: NFPA-805 is discussed as being skewed in that the NRC staff requires inputs to be biased conservatively and uncertainties to be high, this skewing results and increasing the importance of fire vs. other scenarios. Yesterday, we heard similar concerns with the Vogtle level III effort. Is the NRC concerned that best estimate values are not being used? Is there too much input from folks still thinking deterministically instead of [with] a risk-informed mindset?

Answer 18 [response from Hossein Hamzehee, NRC]: In reviewing licensees' applications to adopt NFPA 805, the NRC staff uses state-of-the-art methods, tools, and data for the conduct of a fire PRA. These methods have been developed through joint activities between the industry and the NRC. Industry participants supported analyses and provided peer review of the methods. NRC processes are available for the staff to consider new information that becomes available from ongoing joint activities between the NRC and the industry, to make appropriate changes in the review process, and to replace possible conservatisms in existing methods by more realistic methods and approaches, as warranted. Reviewing a large number of applications in accordance with the structured framework for conducting the overall fire PRA analysis has provided important and valuable risk insights, as implementation of risk-informed approaches intended, and demonstrated that the fire scenarios, in general, are significant contributors to overall risk of nuclear power plants. The NRC staff continues to improve the state of the art in PRA methods through collaboration with the industry to support the use of PRA in regulatory matters, which complements the NRC's deterministic approach.

Question 19: Are the risk-informed tech spec initiatives technology-neutral? Are digital systems modeled with failure probabilities in addition to safety significance?

Answer 19 [response from Hossein Hamzehee, NRC]: Yes, the risk-informed tech spec initiatives are technology-neutral.

Regarding digital I&C:

Digital Instrumentation and Control Probabilistic Risk Assessment - The objective of this research is to identify and develop methods, analytical tools, and regulatory guidance for (1) including models of digital systems in nuclear power plant PRAs, and (2) incorporating digital systems in the NRC's risk-informed licensing and oversight activities.

Research Approach - The NRC has been investigating reliability modeling of digital systems, which encompasses both hardware and software, for several years. Previous projects identified a set of desirable characteristics for reliability models of digital systems and assessed candidate methods against these attributes. In the area of digital hardware reliability, a simulation-based tool has been developed to determine the combinations and sequencing of component level failures that could impact system functions. Current research efforts are focused on developing methods for quantifying software reliability. As an initial step in this area, an expert panel was convened to establish a philosophical basis for modeling software failures in a reliability model. After reviewing several quantitative software reliability methods, two methods to apply to an example software-based protection system in a proof-of-concept study: the Bayesian Belief Network (BBN) approach and the statistical testing method. These methods are being applied to the Loop Operating Control System (LOCS) of the Idaho National Laboratory (INL) Advanced Testing Reactor (ATR). The work has highlighted several areas for additional research for PRA modeling of digital systems is needed, including the following:

- Defining and identifying failure modes of digital systems and determining the effects of their combinations on the system.
- Methods and parameter data for modeling self-diagnostics, reconfiguration, and surveillance, including using other components to detect failures.
- Data on hardware failures of digital components, including addressing the potential issue of double-crediting fault-tolerant features, such as self-diagnostics.
- Data and methods for modeling common-cause failures (CCFs) of digital components.
- Methods for addressing human reliability and modeling uncertainties in modeling digital systems.

Even if an acceptable method is established for modeling digital systems in a PRA and progress is made in the above areas, (1) the level of effort and expertise required to develop and quantify the models will need to be practical for vendors and licensees, and (2) the level of uncertainty associated with the quantitative results will need to be sufficiently constrained so that the results are useful for regulatory applications. Therefore, a goal of this research program is to assess the practicality and usefulness of including digital systems in nuclear plant PRAs.

Recent accomplishments and near term objectives include the following:

- Completed development of a failure mode taxonomy for digital I&C system performed by the OECD/NEA Working Group on Risk Assessment (WGRISK) (NEA/CSNI/R(2014)16, "Failure Modes Taxonomy for Reliability Assessment of Digital I&C Systems for PRA".)
- In collaboration with the Korea Atomic Energy Research Institute, work is ongoing to quantify software reliability using BBN-based on software development cycle quality attributes.
- Ongoing work to estimate the reliability, including software, of the ATR LOCS using PRA-based statistical testing.

For More Information Contact Ming Ling, RES/DRA, at Ming.Li@nrg.gov.

Question 20: Could information transmitted to the iPhone and iPad using the PRA Insights App be proprietary or even security-related? If so, are these cyber security implications?

Answer 20 [response from Greg Krueger, Exelon]: The "App" is essentially a shortcut to an internal Intranet drive that contains the PRA information. It works when a company laptop or mobile device is within the wireless network present in Exelon company locations. Access to the drive and link to a mobile device requires approved computer access and mobile device registration.

Question 21: If a lot of individuals have information on the risk at the facilities easily accessible, it can be imagined that there is also an increase in malevolent action risk. Is that managed, or is it negligible?

Answer 21 [response from Greg Krueger, Exelon]: The "App" is essentially a shortcut to an internal Intranet drive that contains the PRA information. It works when a company laptop or mobile device is within the wireless network present in Exelon company locations. Access to the drive and link to a mobile device requires approved computer access and mobile device registration.

Question 22: One slide notes "Risk Informing the Maintenance Rule". Isn't this already a risk-informed program? What are you doing differently?

Answer 22 [response from Greg Krueger, Exelon]: The slide in Question is intending to show actions that have been taken or are related to the different hazards. As noted, the Maintenance Rule is already risk-informed. This was an attempt to show that the internal events PRA provides the basis for risk informing the Maintenance Rule whereas the other hazards typically do not. Perhaps it would have been more appropriate to just cite the Maintenance Rule rather than characterizing that the implementation had another risk overlay beyond what the Rule already requires.

Question 23: Given most of the Exelon fleet remains mostly deterministic for fire protection (non-805), how does it justify "technically" (not legally) adhering to an outdated licensing basis of "one spurious operation per fire"? If PRA is such a powerful tool, why not transition to something like NFPA 805?

Answer 23 [response from Greg Krueger, Exelon]: The fire PRAs are a powerful tool and are being developed independent of the licensing basis. Implementation of a voluntary process to highlight specific fire areas, using the insights from the fire PRAs, recognizes that fire risk can be impacted and mitigated through knowledge and changing expectations. This voluntary internal initiative goes beyond the deterministic or risk-informed license requirements to address fire risk insights.

Question 24: Given Exelon's partnership with AMEC to explore international opportunities, what risk challenges do you foresee?

Answer 24 [response from Greg Krueger, Exelon]: I am not aware of such a partnership; therefore I believe it would be inappropriate to comment or speculate.

TECHNICAL SESSIONS
Wednesday, March 11, 2015, 1:30 p.m. – 3:00 p.m.

W17 Gas Accumulation and Management: Remaining Issues and their Resolution

Session Chair: Tim McGinty, Director, Division of Safety Systems, NRR/NRC, 301-415-3283, Tim.McGinty@nrc.gov

Session Coordinator: Jennifer Whitman, Reactor Systems Engineer, Division of Safety Systems, NRR/NRC, 301-415-3253, Jennifer.Whitman@nrc.gov

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

W18 Implementation of Lessons Learned from the Fukushima Dai-ichi Accident

Session Chair: William M. Dean, Director, Office of Nuclear Reactor Regulation, NRC, 301-415-1270, Bill.Dean@nrc.gov

Session Co-Coordinators: Kevin Witt, Project Manager, Japan Lessons Learned Project Directorate, NRR/NRC, 301-415-2145, Kevin.Witt@nrc.gov

Jon Hopkins, Senior Project Manager, Division of Inspection and Regional Support, NRR/NRC, 301-415-3027, Jon.Hopkins@nrc.gov

The questions below were not answered during the above session.

Question 1 [addressed to all speakers]: What is your view of differences between counties in responding to the Fukushima accident?

Answer 1 (response from Michael Johnson, NRC]: Overall, regulatory actions taken in response to lessons learned from the Fukushima accident are similar between the U.S. and other countries. While the approaches taken differ from one country to the next, we expect the outcome of our initiatives will result in a similar safety benefit. It is important to note that the high priority lessons learned initiatives being addressed by the international community are consistent with the activities currently being implemented by the Nuclear Regulatory Commission (NRC) in the U.S. The NRC has and will continue to work collaboratively with our international counterparts, to ensure we all fully benefit from each other's insights, perspectives, and activities. Some of the major focus areas that each country is working on in response to Fukushima lessons learned include protection from external hazards, mitigation and prevention of severe accidents, and enhancement of emergency response capabilities.

Answer 1 [response from TANG Bo, China National Nuclear Safety Administration]: My view is some countries conducted more hardware and software improvements, such as China, and other countries emphasized more research. The response to the Fukushima accident between countries depends on different evaluations about the Fukushima accident, differences between nuclear power plants and sites, and resources.

Answer 1 *[*response from Philippe Jamet, Autorité de Sûreté Nucléaire]: At the European level:

- Stress tests conclusions have led some countries to implement technical measures such as filtered vents or hydrogen recombiners; necessity of some of these measures was already shown in previous events (as the Three Mile Island accident)
- Almost all countries have strengthened their dispositions related to the loss of the ultimate heat sink, the loss of electrical power (station blackout situation) or severe accident management (including mobile equipment, etc.)
- On the other hand, level of improvement of fixed equipment varies from one country to another

Across the World, ASN notes that the amount of work carried out or the scale of implemented modifications cover a very wide range from one country to another.

Question 2 [addressed to all speakers]: How has the nuclear safety culture changed since the Fukushima accident?

Answer 2 [response from Michael Johnson, NRC]: The NRC has long recognized the importance of a strong nuclear safety culture. The NRC defines nuclear safety culture as the core values and behaviors resulting from a collective commitment by leaders and individuals to emphasize safety over competing goals to ensure protection of people and the environment. Just shortly after the Fukushima accident, the NRC published a Safety Culture Policy Statement in June 2011. While this policy statement did not specifically address any lessons learned from the Fukushima accident, it communicated the NRC's expectation that individuals and organizations performing regulated activities must establish and maintain a healthy safety culture that recognizes the safety and security significance of their activities and the nature and complexity of their organizations and functions. Since the Fukushima accident, the NRC has highlighted the importance of safety culture and is continuing to educate licensees on how to maintain a strong nuclear safety culture.

Answer 2 [response from TANG Bo, China National Nuclear Safety Administration]: After the Fukushima accident, the NNSA adopted more measures to encourage that the nuclear industry improve safety culture.

Answer 2 [response from Philippe Jamet, Autorité de Sûreté Nucléaire]: To date, no formal process has been initiated related to safety culture. But ASN is participating to various working groups related to safety culture in the framework of the Nuclear Energy Agency or the International Atomic Energy Agency (IAEA).

Moreover, in the framework of the stress tests carried out both at European and national levels, a considerable amount of work has been completed on various topics as assessment of external natural hazards or severe accident management. This work strengthened ASN's opinion that the occurrence of a severe accident, anywhere in the world, cannot be completely ruled out. It has also strengthened ASN's point of view that it is necessary to re-examine on a periodic basis, nuclear installations' safety levels.

A severe accident of this magnitude can occur in any country in the world and lessons learned from the Fukushima accident must be taken into account not only by Japan but by all countries.

Answer 2 [response from Toyoshi Fuketa, Japan Nuclear Regulation Authority]: The Japanese regulator and operators have made various efforts toward safety culture improvement based on lessons learned from Fukushima Daiichi Accident. It is noteworthy that their safety awareness changes entirely. Before the accident, they thought that a severe accident would not occur as long as a plant meets the regulatory requirements. Now they recognize that a severe accident could happen no matter how a plant satisfies regulatory requirements, since it means just achievement of certain level of safety. Continuous improvement of safety becomes mandatory for operators in accordance with this recognition.

Question 3 [addressed to all speakers]: Based on today's knowledge, what would you have done differently following the accident?

Answer 3 [response from Michael Johnson, NRC]: The NRC has made substantial progress in implementing lessons learned from the accident. However, as with any project of this magnitude, in retrospect, there are areas where efficiencies could have been gained. For example, had the complexity and challenges associated with the flood hazard reevaluations been identified earlier, the NRC could have adjusted resources at the onset to ensure the reevaluations remained on schedule (the flood hazard reevaluations are the only Tier 1 activity behind schedule). In such cases, the NRC has proactively looked for way to improve efficiency and effectiveness. In the case of the flooding reevaluations, the NRC is currently evaluating changes to the guidance for conducting these reevaluations to get them back on schedule.

Answer 3 [response from TANG Bo, China National Nuclear Safety Administration]: Based on today's knowledge, emergency operating procedures and severe accident management guidelines should be upgraded continually and provide instruction to operating persons following an accident.

Answer 3 [response from Philippe Jamet, Autorité de Sûreté Nucléaire]: Based on today's knowledge, European stress tests process, associated peer reviews process and first lessons learned, our conclusions are not called into question.

Answer 3 [response from Toyoshi Fuketa, Japan Nuclear Regulation Authority]: We should have been more agile and could have done much more to pursue continuous improvement.

Question 4 [addressed to all speakers]: How were extreme conditions defined/selected for the stress tests performed to assess your nuclear plant capabilities? Were they different for different locations?

Answer 4 [response from Michael Johnson, NRC]: The stress tests were part of a review of European Union (EU) nuclear plants on the basis of a comprehensive and transparent risk and safety assessment in light of the lessons learned from the Fukushima accident. The European Nuclear Safety Regulatory Group (ENSREG) and the European Commission developed the scope and content of the tests. The stress tests were defined as a targeted reassessment of the safety margins for different nuclear power plants in light of the lessons learned from the events in Fukushima. Each test evaluated how the nuclear power plant would respond to site-specific extreme natural events challenging the plant's safety functions. Multiple sites in different European locations participated in this assessment and conditions for their analyses were selected based on their design and environmental hazards given their location.

Answer 4 [response from TANG Bo, China National Nuclear Safety Administration]: For different sites, we selected similar hazard conditions for stress tests, e.g. 1.5 times of safety shutdown earthquake for seismic margin analysis, anti-flooding margin evaluation by increasing water level gradually until core damage, and then station blackout.

Answer 4 [response from Philippe Jamet, Autorité de Sûreté Nucléaire]: Stress tests analyses were focused on safety margins and robustness of nuclear installations regarding beyond design extreme natural events (earthquake, flooding, etc.). Earthquake and flooding events taken into account for the design of nuclear installations were site-specific.

Answer 4 [response from Toyoshi Fuketa, Japan Nuclear Regulation Authority]: In the stress test done by the NISA, the regulatory body at that time, the design basis earthquake and tsunami at that time were used. Obviously, therefore, they were different for different site locations.

Question 5 [addressed to all speakers]: How important for safety is installing external filters on vents?

Answer 5 [response from Michael Johnson, NRC]: The NRC previously completed an evaluation (Commission Paper SECY-12-0157) of whether there should be new requirements for nuclear power plants to add filters to containment vents. The evaluation considered various costs and benefits associated with requiring containment vents, as well as the considerations of factors that are difficult to present in quantitative measures, such as certain societal impacts. As directed by the Commission, the NRC staff is currently conducting a rulemaking to enhance capabilities to maintain containment integrity and to cool core debris. As part of that rulemaking activity, the NRC is evaluating the need for additional requirements associated with filtration or confinement of radioactive materials that may be released following core damage. The staff intends to provide the Commission with a paper in spring 2015 describing the preliminary results of the analysis and the path forward. Additional information regarding the progress of NRC lessons learned activities can be found at the Japan Lessons Learned website.

Answer 5 [response from TANG Bo, China National Nuclear Safety Administration]: In China, containment filtered vents have been installed in the CPR1000 series nuclear power plants, the purpose of which is to maintain the integrity of containment in case of a severe accident. But one must be careful when evaluating the containment filtered vent. First, it is a very serious condition to consider the radiological impact of the environment and public when running with a containment filtered vent. Second, containment is a potential vacuum so it can lose integrity due to cooldown in long term.

Answer 5 [response from Michael Johnson, NRC]: Installation of containment filtered vents is firstly important for reactors for which a severe accident was not taken into account during the design phase (such as Generation II reactors). In the event of a severe accident, such equipment is enabled to limit radioactive releases and also to partially control radioactive releases. This could give the operator and organizations in charge of public protection the possibility to coordinate their decisions.

Answer 5 [answered by Toyoshi Fuketa, Japan Nuclear Regulation Authority]: Our new regulatory requirements are performance-based. The filtered containment venting is one of the acceptable measures that licensees may take in order to prevent containment failure due to high pressure and/or high temperature.

Question 6 [addressed to Michael Johnson, NRC]: Please, could you clarify the NRC position concerning the safety objective of “avoiding massive releases”? What payout, equipment modifications are required to reach this target, either for PWR or BWR, by NRC?

Answer 6 [response from Michael Johnson, NRC]: The mission of the NRC is to license and regulate the civilian use of radioactive materials to protect public health and safety, promote the common defense and security, and protect the environment. As such, the agency's safety objective is to prevent unplanned radioactive releases to the environment regardless of their magnitude or size. The agency ensures that large releases of radioactive material are avoided through both preventive and mitigative measures. In light of the accident at Fukushima, the NRC has imposed additional requirements to enable licensees to better respond to beyond-design-basis events and maintain key safety functions. These enhancements include a combination of equipment modifications, improvements to emergency response and emergency preparedness, as well as reevaluations of external natural hazards.

The NRC's post-Fukushima regulatory requirements were generally promulgated to provide reasonable assurance of adequate protection of public health and safety, which are pursued without consideration of costs. The NRC, however, works with its licensees and applicants to find a cost-effective means of implementing requirements imposed for adequate protection purposes. The NRC can also implement regulatory requirements to substantially enhance safety beyond those that are needed for reasonable assurance of adequate protection; these types of safety enhancements must be cost-justified. An example of a post-Fukushima cost-justified safety enhancement was the addition of a requirement that the reliable hardened vents initially required by Order EA-12-050 be severe accident capable (Order EA-13-109 superseded the original order and imposed this additional requirement).

Additional information on the type of modifications being implemented at U.S. nuclear power plants can be accessed from Implementation Status page on the NRC's Japan Lessons Learned website at http://www.nrc.gov/reactors/operating/ops-experience/japan-dashboard/japan-plants.html.

Question 7 [addressed to Michael Johnson, NRC]: Considering the potential for multi-unit sites to experience multi-unit events, should the design basis accident regulations be modified to require analysis of these classes of accidents for siting/licensing?

Answer 7 [response from Michael Johnson, NRC]: A number of the NRC's post-Fukushima safety enhancements have strengthened licensees' abilities to respond to multi-unit events. For example, in the implementation of the mitigation strategies order (Order EA-12-049), the NRC established a requirement for licensees to procure and maintain N+1 pieces of equipment, where "N" is the number of units on-site. Thus, a two-unit site would nominally have at least three portable pumps, three sets of portable ac/dc power supplies, three sets of hoses and cables, etc. Licensees need to demonstrate that they have sufficient staffing to implement these strategies simultaneously for all the units at a site. Additionally, as part of the NRC's Fukushima lessons learned activities, the NRC has ensured that licensees have the capability to perform a multiunit radiological dose assessment using the licensee's site-specific dose assessment software and approach. Since the NRC's ongoing Fukushima lessons learned regulatory activities address these types of considerations, the NRC does not plan to modify any design basis regulations.

Question 8 [addressed to Michael Johnson, NRC]: What's your perspective on how much of the beyond design basis requirements will make it into the plant's design basis?

Answer 8 [response from Michael Johnson, NRC]: The NRC staff recently provided a paper to the Commission (COMSECY-14-0037) regarding the integration of mitigating strategies for beyond-design-basis external events and the reevaluation of flooding hazards. The Commission affirmed that mitigating strategies will need to address the reevaluated flooding hazards, which means that the flooding reevaluations would be used to define functional requirements and reference bounds for those specific structures, systems, and components (SSCs) used to support key safety functions within the mitigating strategies for beyond-design-basis external events. The NRC staff will determine if flooding or other external hazards warrant possible regulatory action beyond required implementation of mitigating strategies on a plant-specific basis if justified by evaluations performed in accordance with 10 CFR 50.109, "Backfitting." An example where such action might be considered is a flooding scenario with a relatively high estimated frequency and an associated high probability of the flooding event leading to core damage. In such a case, the NRC staff may find that reliance on mitigating strategies alone is not sufficient and additional flood protection or mitigation requirements may be warranted. The Commission directed the staff to allow flexibility in the way in which licensees address such potential vulnerabilities, including providing licensees the opportunity to demonstrate that vulnerabilities identified may be less risk significant when more realistic assumptions are applied in the analyses.

Question 9 [addressed to Michael Johnson, NRC]: Regarding the NRC's Near Term Task Force Recommendation 2.1 on seismic reevaluations, is the NRC planning to form an intra-agency team of staff from the Office of New Reactors, the Office of Nuclear Reactor Regulation, and the Office of Nuclear Regulatory Research to review future seismic probabilistic risk assessments (SPRAs)? Which NRC department will have the lead role for this process?

Answer 9 [response from Michael Johnson, NRC]: The NRC plans to use agency-wide technical experts and resources to review SPRAs and other detailed seismic submittals. Staff from the Office of Nuclear Reactor Regulation will have the lead role in overseeing effective progress of the reviews, with technical support provided by the Offices of New Reactors and Nuclear Regulatory Research.

Question 10 [addressed to Michael Johnson, NRC]: The Fukushima accident was arguably rooted in failure to respond to new information that the seismic/tsunami hazard was much greater than the Fukushima design basis; as this was known many years prior to 2011. What has been done to strengthen the regulatory process for responding to new information?

Answer 10 [response from Michael Johnson, NRC]: One of the NRC's ongoing Fukushima lessons learned activities is the reevaluation of licensee's seismic and flooding hazards using modern techniques and updated information. The licensees of operating nuclear power plants are in the process of reevaluating their seismic and flooding hazards and, if appropriate, have provided interim actions to protect the site against the updated hazard. The licensees are required to perform more detailed assessments to further identify and address vulnerabilities. Also, the NRC is implementing a requirement for U.S. nuclear power plants to implement strategies to keep the reactor core and spent fuel pool cool, as well as to protect the reactor's containment, following an extreme external event beyond the design basis of the facility. Additional information regarding the progress of NRC lessons learned activities can be found at the Japan Lessons Learned website.

Through our existing regulatory oversight and research programs, the NRC regularly reviews new information on potential earthquake and flooding hazards, along with other potential challenges to nuclear safety. For example, before the Fukushima accident occurred, the NRC was in the process of examining updated seismic hazard information developed by the U.S. Geological Survey in 2008 to assess potential safety implications for nuclear power plants in central and eastern U.S. A similar process was followed for the issue of flooding caused by upstream dam failures, where the NRC had identified this as a generic safety issue before the Fukushima accident and now the NRC is taking extensive action through the lessons learned process. The NRC has confidence that our research and regulatory programs will promptly identify safety issues before they have an impact on public health and safety.

Question 11 [addressed to Michael Johnson, NRC]: This year the U.S. will have its first national exercise involving an nuclear power plant in 15 years. What will/can NRC leadership do to stress the need for more frequent (~5 years) national exercises involving a nuclear or radiological event?

Answer 11 [response from Michael Johnson, NRC]: In response to the Fukushima accident, the NRC is has either taken action or has actions planned to enhance response to a nuclear or radiological event. For example, as recommended by Near Term Task Force (NTTF) Recommendation 11.2, the NRC is planning to work with the Federal Emergency Management Agency (FEMA), States, and other external stakeholders to evaluate insights from the Fukushima accident to identify potential enhancements to the U.S. decision-making framework, including the concepts of recovery and reentry. These enhancements will enable Federal, State, and local authorities to more effectively respond to a radiological incident. The NRC is also working to strengthen requirements and procedures for on-site emergency preparedness in response to lessons learned from the Fukushima Dai-ichi accident. As part of the NRC's implementation of the mitigation strategies order (EA-12-049), the NRC is ensuring that licensee's include periodic training and exercises for multi-unit and prolonged station blackout (SBO) scenarios and to practice (simulate) the identification and acquisition of offsite resources, as recommended by the NRC's NTTF Recommendation 9.1. Additional information regarding the progress of NRC lessons learned activities can be found at the Japan Lessons Learned website.

Question 12 [addressed to Michael Johnson, NRC]: Does the NRC have access to detail design information (drawings, specs document) onsite for use during an emergency? Do other Regulators store such information (copies) rather than relying on plants or utilities?

Answer 12 [response from Michael Johnson, NRC]: The NRC maintains extensive information for all regulated nuclear facilities in the U.S., most of which is accessible via the NRC's Agencywide Documents Access and Management System. This information, along with emergency communication capabilities provided by the NRC's Emergency Operations Center, enables the NRC staff to provide oversight of any U.S. nuclear power plant during an emergency. In response to the Fukushima accident, the NRC and industry have enhanced emergency response capabilities to ensure that loss of power will not compromise communications on-site and off-site. These measures provide additional assurances that the NRC will be able to communicate and obtain accident data from the sites during an emergency situation.

Question 13 [addressed to Michael Johnson, NRC]: On communication and power supply/SBO issues, if the cell towers fail and cell phones are lost during SBO, do plants have independently-powered cell phones?

Answer 13 [response from Michael Johnson, NRC]: In response to the Fukushima accident, the NRC required that all nuclear power plant licensees assess a large-scale event that (1) causes the loss of all alternating current power, (2) affects all units at their site, and (3) impedes access to the site. This letter also required that licensees assess their means to power communications equipment onsite and offsite during a prolonged station blackout event and to assess and carry out enhancements to help ensure that communications can be maintained during such an event. The assessment criteria assumed that the communications infrastructure was damaged in a 25-mile radius around the site, including cell towers, which renders cell phones out of service. Licensees have responded to this request with a list of the communications equipment that they have procured for this situation, including various systems for on-site communications and the use of satellite phones for communication with offsite facilities, local government organizations, and the NRC. Additional information regarding the progress of NRC lessons learned activities can be found at the Japan Lessons Learned website.

Question 14 [addressed to Michael Johnson, NRC]: What specific lessons were learned regarding design basis loss of coolant accident progression (not severe accident or beyond design basis)? Will assumptions of gap release or melt release timing be adjusted?

Answer 14 [response from Michael Johnson, NRC]: At this time, the NRC is not aware of specific lessons learned from Fukushima on loss of coolant accident progression. However, the NRC is continuing to work with the Government of Japan and other international partners to learn more about how the accident progressed. Additionally, the International Atomic Energy Agency and the Nuclear Energy Agency are continuously evaluating the situation in Japan and around the world to determine if any further safety improvements should be implemented. The NRC is maintaining cognizance of these activities and if any new information is discovered, the NRC will evaluate it for applicability to our existing regulatory oversight and research programs.

Question 15 [addressed to TANG Bo, China National Nuclear Safety Administration]: How does China plan to expand your regulatory capacity under conditions of rigid nuclear power growth?

Answer 15 [response from TANG Bo, China National Nuclear Safety Administration]: Following the growth of nuclear power, the nuclear regulatory capacity of China has been expanded continually. After the Fukushima accident, the human and financial resource has been increased again. For example, staff have been added from about 50 persons to 80 persons in NNSA, and from 120 to 600 in NSC (Nuclear Safety Center), which is the technical support organization (TSO) of NNSA, and from about 100 to 300 in regional office. Recently the government of China has given approval to establish a laboratory for research and experiment of nuclear safety technology. I believe NNSA will further expand following the nuclear power growth of China.

Question 16 [addressed to TANG Bo, China National Nuclear Safety Administration]: What is the design for the new reactor at Hongyanhe? Will the reactors have to address Fukushima lessons learned? Does NNSA have sufficient staff to review the applications of the new reactors?

Answer 16 [response from TANG Bo, China National Nuclear Safety Administration]: The new reactor at Hongyanhe, Units 5 and 6, is a modified type of CPR1000 reactor. The passive cooling systems for secondary side and passive water supplement for auxiliary feedwater tank, IVR, etc, have been incorporated into the design. Of course, NNSA requirements for improvement after Fukushima have been addressed. We know it is a great challenge for NNSA to review the applications of new reactors, so we have adopted a series of measures, for example, standard design review, and inviting other organizations to join in the review process, etc.

Question 17 [addressed to TANG Bo, China National Nuclear Safety Administration]: Does China plan to impose more comprehensive fire safety regulations similar to the U.S. regulations?

Answer 17 [response from TANG Bo, China National Nuclear Safety Administration]: In general, Chinese safety regulations about fire protection refer to IAEA safety standards, but due to lack of detailed standards and codes on fire protection of nuclear power plants in China, we refer to many foreign standards and codes, such as regulatory guides and national fire protection association (NFPA) standards of the U.S. and RCC-I of France.

Question 18 [addressed to TANG Bo, China National Nuclear Safety Administration]: Is mobile equipment stored in structures designed for flood and earthquake hazard greater than design basis such as 1.5 times the design basis for earthquake?

Answer 18 [response from TANG Bo, China National Nuclear Safety Administration]: No. We required the structures which store mobile equipment were checked against the safe shutdown earthquake (SSE).

Question 19 [addressed to TANG Bo, China National Nuclear Safety Administration]: What is the basis for 6-hour design capacity for mobile equipment for NPPs including AP-1000?

Answer 19 [response from TANG Bo, China National Nuclear Safety Administration]: The design capacity for mobile equipment is determined against 6-hours residual heat after reactor shutdown, for which we consider several factors:

1. The exercises show that operators need 2~4hours to install, connect and start the mobile equipment;
2. The capacity of auxiliary feedwater tank can provide 6 hours feedwater for steam generator;
3. According to "generic technical guideline" of NNSA, after implementing water-proof seal, one train of residual heat removal (auxiliary feedwater) should be ensured to function within 6 hours at a water level of design basis flood level superposed with a precipitation level of once in 1000 years.

The evaluation above is based on operating NPPs in China, those are traditional designs. For AP-1000, other evaluations are necessary.

Question 20 [addressed to TANG Bo, China National Nuclear Safety Administration]: Could China describe its requirements for emergency evacuation planning and Potassium Iodide (KI) distribution? Were any of these requirements revised due to the Fukushima accident?

Answer 20 [response from TANG Bo, China National Nuclear Safety Administration]: In China, on-site emergency plans will be implemented by the nuclear power plant operating organization in case of serious accident, and off-site emergency plans will be implemented by the local government. There are emergency evacuation and KI distribution arrangements in both the on-site and off-site emergency plans. After the Fukushima accident, the emergency plan has conducted some improvements, for example, environment monitor, inhabitability of emergency control center, etc.

Question 21 [addressed to Philippe Jamet, Autorité de Sûreté Nucléaire]: Canada and U.S. said that substantive amendments would be in place by 2016. France said that it would take longer. Why is there a difference in schedule?

Answer 21 [response from Philippe Jamet, Autorité de Sûreté Nucléaire]: Due to the significant improvements required by ASN, full implementation will take a few years. For example, it has been asked to EDF to add for each reactor (58 operating reactors in France) an additional ultimate electricity generator (3.5 MegaWatts-electric each) which can withstand extreme natural hazards. Manufacturing such equipment requires time. In the meantime, a set of temporary or mobile measures has been implemented to reinforce protection against transient situations of total loss of the heat sink or electrical power supplies: These measures include, for example, the installation of medium-power generator sets on each reactor, the reinforcing of the local emergency response means (pumps, generator sets, hoses, etc.).

The schedule of implementation of the measures resulting from the stress test process in France is as follows:

- The first phase from the accident up to 2014-2015 covers the definition of the hardened safety core, the implementation of nuclear rapid response force and transitory measures, for example, one diesel generator is added to each reactor or improvement regarding seismic.
- The second phase up to 2018-2020 covers the implementation of a large part of the hardened safety core equipment, mainly the ultimate diesel generator and the ultimate water make-up system, one for each reactor, the bunkered emergency crisis center, one for each site. All the equipment has to be designed to withstand extreme natural hazards.
- The third phase covers remaining modifications. The deadlines are still under discussions between the licensee and ASN.

Question 22 [addressed to Philippe Jamet, Autorité de Sûreté Nucléaire]: Has there been an effort to benchmark the Stress Tests against the requirements of those European countries that did not participate in the Stress Tests?

Answer 22 [response from Philippe Jamet, Autorité de Sûreté Nucléaire]: Stress tests carried out by European countries following the Fukushima accident used common terms of reference and common specifications. 17 European countries are involved in this on-going process (list available at http://www.ensreg.eu/EU-Stress-Tests/Country-Specific-Reports/EU-Member-States) and regulators from Turkey, Switzerland and Ukraine are involved as observers. Specifications have been drafted by WENRA (Western European Nuclear Regulators Association - http://www.wenra.org/) and adopted by ENSREG (http://www.ensreg.eu/) and the European Commission.

The European approach includes benchmark and peer reviews at various steps of the process:

- A peer review has been carried out about assessment of nuclear installations safety levels and technical measures proposed to increase safety levels,
- A peer review has been carried out about national action plans proposed by each European country (each country had to publish an action plan that reviews the state of implementation of the recommendations resulting from the European stress-tests conducted in 2011 and, more generally, all the further actions decided on these assessments),
- A peer review is carried out about the implementation of these action plans.

Conclusions of these peer reviews are public. In April 2013, an international seminar was held in Brussels (Belgium) aimed to exchange on implementation of post-Fukushima actions at the European level. A new seminar will be held in April 2015 (ENSREG 2nd National Action Plan Workshop). The USNRC will be invited. Additional information is available: http://www.ensreg.eu/EU-Stress-Tests/Follow-up

The Western European Nuclear Regulators Association (WENRA) is composed of the heads of nuclear regulatory bodies from 17 countries. The main objectives of WENRA are to develop a common approach to nuclear safety and to provide an independent capability to examine nuclear safety in applicant countries.

ENSREG is an independent, authoritative expert body created in 2007 following a decision of the European Commission. It is composed of senior officials from the national nuclear safety, radioactive waste safety or radiation protection regulatory authorities and senior civil servants with competence in these fields from all 28 Member States in the European Union and representatives of the European Commission.

Question 23 [addressed to Philippe Jamet, Autorité de Sûreté Nucléaire]: Could France elaborate on the Site Emergency Management Center? Does every French NPP have this kind of Center? Who staffs the Center?

Answer 23 [response from Philippe Jamet, Autorité de Sûreté Nucléaire]: Before the Fukushima accident, all French NPPs have an on-site emergency management center. In the light of the Fukushima event, the utility has been required by the regulator to build on each site a new emergency management center which will have to withstand extreme natural hazards. These premises will have to be accessible and habitable at all times and during long-duration emergency situations. They will have a 100 people capacity and could be staffed by the plant's staff of by FARN teams.

Question 24 [addressed to Philippe Jamet, Autorité de Sûreté Nucléaire]: How does the Rapid Response Force system compare to the two U.S. National Response Centers?

Answer 24 [response from Philippe Jamet, Autorité de Sûreté Nucléaire]: The "FARN" (Nuclear Rapid Intervention Force) is a French national emergency arrangement combining specialized crews and equipment able to intervene in less than 24 hours on a site affected by an accident. These teams will be able to back up the teams of the affected NPP and to bring mobile equipment to ensure the makeup of water and provide electrical power. In this context, several modifications have been applied to the reactors to facilitate the connection of the equipment brought by the FARN. This system will provide the capacity to assist a site with 6 accident-stricken reactors.

Question 25 [addressed to Toyoshi Fuketa, Japan Nuclear Regulation Authority]: What are your views on requiring a Level 3 PRA (Probabilistic Risk Assessment)?

Answer 25 [response from Toyoshi Fuketa, Japan Nuclear Regulation Authority]: The Level 3 PRA is one of the useful methods in order to evaluate the effectiveness of protective actions in case of emergency. We are in a position to encourage the licensees to use it. It is, however, not considered as part of regulatory requirement at the moment.

Question 26 [addressed to Toyoshi Fuketa, Japan Nuclear Regulation Authority]: Have the new more strict requirements been applied to all nuclear power plants in Japan?

Answer 26 [response from Toyoshi Fuketa, Japan Nuclear Regulation Authority]: Yes.

Question 27 [addressed to Toyoshi Fuketa, Japan Nuclear Regulation Authority]: How many Nuclear Power Plants (NPPs) in Japan are in operation and how many are shutdown?

Answer 27 [response from Toyoshi Fuketa, Japan Nuclear Regulation Authority]: A total of 54 NPPs were in service at the time of the Fukushima Daiichi accident. After the accident, several licensees decided to close 11 NPPs permanently including the Fukushima Daiichi units 1 to 4. Consequently, as of June 2015, a total of 43 NPPs are in service and all of them are in shutdown.

Question 28 [addressed to Toyoshi Fuketa, Japan Nuclear Regulation Authority]: How are the Japan Nuclear Safety Institute (JANSI) and the NRA working together to improve nuclear safety?

Answer 28 [response from Toyoshi Fuketa, Japan Nuclear Regulation Authority]: The NRA expresses its expectation on JANSI's roles to play and tries to well understand the operator's efforts for improving safety through communication with JANSI.

Question 29 [addressed to Toyoshi Fuketa, Japan Nuclear Regulation Authority]: Can Japan explain the use of "expert judgment" as one source of re-evaluating hazards? How is expert judgment quantified and why is it important to consider in addition to historical records and PRA?

Answer 29 [response from Toyoshi Fuketa, Japan Nuclear Regulation Authority]: Since uncertainty is large in hazard assessments, expert judgment on uncertainties, in addition to historical records evaluation is indispensable. For maintaining appropriate safety margin, various expert opinions have to be examined carefully especially in the case where these opinions vary.

Question 30 [addressed to Toyoshi Fuketa, Japan Nuclear Regulation Authority]: What lessons have been learned in comparing Fukushima with Onagawa NPP site?

Answer 30 [response from Toyoshi Fuketa, Japan Nuclear Regulation Authority]: Onagawa Nuclear Power Station was not severely damaged by the tsunami. Although the tsunami exceeded its design basis tsunami height, it did not reach the site ground level. The NRA strengthened the design bases for natural phenomena in order to prevent simultaneous loss of safety functions due to common cause failure.

Question 31 [addressed to Toyoshi Fuketa, Japan Nuclear Regulation Authority]: Where do Japan NPPs stand in addressing the new strict fire safety regulations?

Answer 31 [response from Toyoshi Fuketa, Japan Nuclear Regulation Authority]: According to the Nuclear Regulation Act amended in 2013, all the NPPs shall be in compliance with the NRA's regulatory requirements, including those for fire protection. Then all the Japanese NPPs that aim to restart must meet the new regulatory requirements.

W19 Leveraging Regional Partnerships for Improved Nuclear Safety and Security Practices

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Session Coordinator: Leah Salisbury, International Relations Specialist, OIP/NRC, 301-415-2585, Leah.Salisbury@nrc.gov

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

W20 Research Efforts Affecting Spent Fuel Storage and Transportation

Session Chair: Meraj Rahimi, Branch Chief, Division of Spent Fuel Management, NMSS/NRC, 301-287-9233, Meraj.Rahimi@nrc.gov

Session Coordinator: Jeremy Smith, Senior Nuclear Engineer, Division of Spent Fuel Management, NMSS/NRC, 301-287-0928, Jeremy.Smith@nrc.gov

The questions below were not answered during the above session.

Question 1 [addressed to Meraj Rahimi, NRC]: What fraction of the research agenda would not be necessary if there was an operating disposal facility in the next 10 years?

Answer 1 [response from Meraj Rahimi, NRC]: If a geologic repository for spent fuel disposal were to begin operating in the next 10 years, much of the current and expected inventory of spent fuel would need to remain in storage for many decades before it could be moved to such a facility. Nearly all of the NRC's current research efforts have some applicability for this projected period of storage and subsequent transportation.

Question 2 [addressed to Meraj Rahimi, NRC]: Can you quantify in ($) how much this R&D agenda costs because of a lack of a repository for disposal?

Answer 2 [response from Meraj Rahimi, NRC]: As stated in the previous response, nearly all of NRC's current storage and research activities have some applicability to the anticipated needs for continued storage, even if a repository were to become available in the next 10 years. The staff has not estimated its specific research costs under different scenarios.

Question 3 [addressed to Meraj Rahimi, NRC]: Do you see the growth of friction stir welding (FSW) having an effect on dry storage (for example, improved welds extending the life of storage containers)?

Answer 3 [response from Meraj Rahimi, NRC]: The FSW that NRC has approved is for welding basket plates inside casks. The NRC has not found evaluated the potential benefits of FSW on storage containers. For any welding process to be used on the pressure or containment boundary of a spent fuel storage system, the process needs to be qualified and approved for use in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (B&PV) and endorsed by the NRC. At this time, FSW is not approved for use for pressure or containment boundary in the ASME B&PV.

Question 4 [addressed to Meraj Rahimi, NRC]: What fraction of the research agenda would not be necessary if there was an operating disposal facility in the next 10 years?

Answer 4 [response from Meraj Rahimi, NRC]: If a geologic repository for spent fuel disposal were to begin operating in the next 10 years, much of the current and expected inventory of spent fuel would need to remain in storage for many decades before it could be moved to such a facility. Nearly all of the NRC's current research efforts have some applicability for this projected period of storage and subsequent transportation.

Question 5 [addressed to Meraj Rahimi, NRC]: Can you quantify in ($) how much this R&D agenda costs because of a lack of a repository for disposal?

Answer 5 [response from Meraj Rahimi, NRC]: As stated in the previous response, nearly all of NRC's current storage and research activities have some applicability to the anticipated needs for continued storage, even if a repository were to become available in the next 10 years. The staff has not estimated its specific research costs under different scenarios.

Question 6 [addressed to Meraj Rahimi, NRC]: Do you see the growth of friction stir welding (FSW) having an effect on dry storage (for example, improved welds extending the life of storage containers)?

Answer 6 [response from Meraj Rahimi, NRC]: The FSW that NRC has approved is for welding basket plates inside casks. The NRC has not found evaluated the potential benefits of FSW on storage containers. For any welding process to be used on the pressure or containment boundary of a spent fuel storage system, the process needs to be qualified and approved for use in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (B&PV) and endorsed by the NRC. At this time, FSW is not approved for use for pressure or containment boundary in the ASME B&PV.

Question 7 [addressed to Gordon Bjorkman, NRC]: When can the NRC results be considered for licensing high burnup fuel for storage and transportation?

Answer 7 [response from Gordon Bjorkman, NRC]: The NRC plans to develop a guidance document on how the results can be used in the licensing process.

Question 8 [addressed to Gordon Bjorkman, NRC]: Good Work!! Are the results directly applicable to 17x17 fuels with a smaller diameter and thinner cladding and higher hydrogen content?

Answer 8 [response from Gordon Bjorkman, NRC]: They may be directly applicable within limits, but additional testing will have to be done to verify this.

Question 9 [addressed to Gordon Bjorkman, NRC]: In the Phase 1 tests at ORNL, cladding was tested to 10^6 or 10^7 cycles? How many cycles would the cladding experience when shipped across the country?

Answer 9 [response from Gordon Bjorkman, NRC]: The vibration (displacement) time history of the rod during transport will be composed of many frequencies. The higher frequencies will exhibit very small displacements and small cladding strains. The lower frequencies will have higher displacements and cladding strains, particularly those frequencies close to the natural frequency of the rod. At higher frequencies there will be more cycles than at lower frequencies. Therefore, the number of cycles experienced by the cladding will depend on the frequency being considered. Fatigue damage laws take all this into account. On a cross country trip 10^6 to 10^7 cycles is expected to bound the response of the highest frequencies that could cause fatigue damage.

Question 10 [addressed to Gordon Bjorkman, NRC]: Based upon the results you have seen so far, are you optimistic that the initial results can be applied to both rail and over-the-road transport? That is, you do not anticipate cladding failure in either transport mode?

Answer 10 [response from Gordon Bjorkman, NRC]: The results are directly applicable to both rail and over-the-road transport.

Question 11 [addressed to Gordon Bjorkman, NRC]: Are there any results for low burnup PWR ZIR-4 used fuel to compare your results to the high burnup fuel? If so, how did the results compare? If not, what differences are expected?

Answer 11 [response from Gordon Bjorkman, NRC]: At the moment there are no results for low burnup fuel. The NRC does not expect to see significant differences in the bending and fatigue response between low and high burnup fuel <u>for the range of strains exhibited in the tests.</u>

Question 12 [addressed to Gordon Bjorkman, NRC]: In the next phase of testing with cladding that has undergone re-orientation, how prototypical will the cladding re-orientation be? Will actual vacuum drying conditions be simulated?

Answer 12 [response from Gordon Bjorkman, NRC]: We do not anticipate that hydride re-orientation will have a significant effect on the results, because the normal tensile bending stresses in the cladding are parallel to the plane of both the circumferential and radial hydrides.

Question 13 [addressed to Gordon Bjorkman, NRC]: You noted that more tests are necessary ... what tests? What do you want to find out from these additional tests?

Answer 13 [response from Gordon Bjorkman, NRC]: We want to test different cladding types, different rod diameters, different levels of hydride re-orientation, PWR fuel, BWR fuel, etc. to be able to fully characterize the bending and fatigue properties of as many different fuel rods as possible.

Question 14 [addressed to Matt Hiser, NRC]: It seems that much of the focus of monitoring was on dry storage in your presentation. Are there any unique monitoring aspects for transportation that you can identify?

Answer 14 [response from Matt Hiser, NRC]: Yes, this effort focused primarily on the extended storage scenario, particularly for degradation modes that generally take long periods of time to manifest. One potential unique monitoring aspect for transportation could be measuring vibration levels during transport to ensure transport loading is within the design basis of the transportation package.

Question 14 [addressed to Matt Hiser, NRC]: With the advances occurring on remote monitoring at Fukishima, are there any items that may be of benefit to this program (e.g. muon detectors proposed for fuel location at Fukishima)?

Answer 14 [response from Matt Hiser, NRC]: This report looked broadly at the available literature fore any relevant information. When this report is updated in the future, we will certainly look at all available information sources, including potential applicability of techniques used at Fukushima.

Question 15 [addressed to Matt Hiser, NRC]: From a safety standpoint, is there a <u>real</u> need for continuous monitoring verses periodic inspection?

Answer 15 [response from Matt Hiser, NRC]: Development of in-service monitoring methods for storage systems and components was identified as a crosscutting need (ML14043A423). NRC staff considers the ability to monitor system components for degradation to be a valuable tool for ensuring continued safety of SNF storage. The purpose of the monitoring report was to provide NMSS staff reviewers with the information necessary to review monitoring techniques potentially proposed by licensees for aging management. This research was undertaken from the perspective of a potential 300-year extended storage period, but could be applicable in nearer timeframes. Assessing the need for continuous monitoring as opposed to periodic inspection at this time or in the future was not within the scope of this project.

Question 16 [addressed to Matt Hiser, NRC]: Does the NRC really consider the CNWRA CISCC test conditions representative of real storage conditions?

Answer 16 [response from Matt Hiser, NRC]: The CNWRA CISCC test conditions are the most representative test data that NRC is aware of. Recent industry inspections of in-service canisters have observed salt concentrations and temperatures in the range of the CNWRA testing within 5 years of cask placement (ML14323A939). NRC recognizes there may be uncertainty in the representativeness of other important factors, such as residual stress. NRC would welcome further industry-sponsored research to assess CISCC susceptibility under "conditions representative of real storage conditions."

W21 Safety Culture Assessments – How is Culture Measured?

Session Chair: Patricia Holahan, Director, Office of Enforcement, NRC, 301-415-2741, Patricia.Holahan@nrc.gov

Session Coordinator: Catherine Thompson, Program Manager, OE/NRC, 301-415-3409, Catherine.Thompson@nrc.gov

The questions below were not answered during the above session.

Question 1 [addressed to NRC]: In what way does the NRC program on safety culture benchmark the results in the nuclear sector with other major industries (pharmaceutical or aerospace for example)?

Answer 1 [response from Molly Keefe, NRR]: The NRC has conducted formal benchmarking and reviews of how other industries and regulatory agencies approach safety culture including, for example, an exchange of safety culture information with the Canadian railway industry. Another example is the literature review conducted by Pacific Northwest National Laboratories available at ADAMS ML13023A054. The NRC staff also regularly participates in informal benchmarking and information exchanges with federal regulators in other industries through interagency roundtable meetings.

Question 2 [addressed to NRC]: Are 12-hour shifts compatible with safety culture and what is NRC doing to make sure workers get enough sleep?

Answer 2 [response from Kamishan Martin, NRR]: The NRC regulations require that licensees schedule individuals who perform functions that are vital to public health in ways which are in agreement with adequate fatigue management. The licensee is responsible for the actual hours worked by these individuals as well as the scheduled hours worked to ensure that individuals are fit for duty and not impaired by chronic or acute fatigue. Furthermore, the NRC requires licensees to have a comprehensive fitness for duty program which addresses fatigue management. If there are safety issues with causal factors possibly extending from an individual not being fit for duty due to fatigue, the Reactor Oversight Program includes provisions for supplemental inspections.

Question 3 [addressed to NRC]: How do nuclear regulatory bodies like the US NRC develop an efficient and an effective safety culture from within? What is, or should be, the role of the Commissioners in this respect?

Answer 3 [response from Susan Salter, OCHCO]: At the US NRC, we strive to create an organizational culture that emphasizes safety, and to create a work environment that encourages all employees and contractors to promptly raise concerns and differing views without fear of reprisal. To support this goal, we work to promote an open collaborative work environment that encourages differing views and opinions to be aired early, and to provide effective feedback on why decisions are made. When differences cannot be resolved, the agency also has formal avenues for raising mission related concerns. Key to a healthy environment for raising concerns and a strong safety culture is leadership commitment and support. The role of the NRC Commissioners is to continue to support and encourage the sharing of different views and opinions.

Question 4 [addressed to NRC]: Please describe NRC's processes for ongoing daily assessment of safety culture performance?

Answer 4 [response from Susan Salter, OCHCO]: Although the NRC does not have a formal, daily process for assessing safety culture, there is close coordination between the Office of the Chief Human Capital Officer (OCHCO), Office of Enforcement (OE), and the Small Business and Civil Rights (SBCR) office, to stay abreast of activity levels concerning EEO complaints, Employee Relations/Labor Relations grievances, and experiences with the agency's Differing Professional Opinions Program and Non-Concurrence Process as early warning indicators of changes in the organizational climate.

The agency does conduct assessments of its safety culture via the Triennial Office of Inspector General (OIG) Safety Culture Climate Survey (SCCS), as well as post-survey assessment activities (focus groups, employee interviews, etc.) In addition, annual government-administered Federal Employee Viewpoint Survey provides an annual check-in on things such as leadership, employee engagement and job satisfaction. Actions plans are developed at both the agency and Office/Region levels to address areas needing improvement and those plans are evaluated each year and updated as necessary.

Question 5 [addressed to NRC]: Given that OIG surveys consistently reveal "weaknesses" in the NRC's internal safety culture, what is being done to fix those "weaknesses" and hasten the agency's safety culture journey?

Answer 5 [response from Susan Salter, OCHCO]: The NRC has a longstanding history of promoting a positive safety culture to ensure the agency achieves its mission. Promptly speaking up and sharing concerns and differing views without fear of negative consequences are key components of our agency safety culture. The last OIG Safety Culture Climate Survey

(SCCS) was conducted in 2012, and while the agency continued to be more favorable than industry and national norms, the overall trend was for less favorable results relative to the 2009 OIG SCCS results. Specific areas for improvement included communicating why decisions were made, addressing negative reactions when using one of the agency's formal programs for raising a concern or different opinion, and recognizing and respecting the value of human differences.

To address these areas, an agency action plan was developed that included the launching of an initiative called "Behavior Matters" which was designed to develop a shared awareness and understanding of the behaviors that support the NRC values. In addition, the agency provided training on "Emotional Intelligence," "Difficult Conversations," and "Civility." To support the agency's environment for raising concerns, assessments were completed and subsequent action plans developed for the agency's "Open Collaborative Work Environment," the Non-Concurrence Process and the Differing Professional Opinions Program. The next SCCS is due to be conducted in the fall of 2015.

Question 6 [addressed to Sonja Haber, Human Performance Analysis Corporation]: During peer review missions such as the IRRS and other peer reviews, experts are asked to review safety culture against IAEA safety standards (requirements). What would be the assessment criteria (i.e. performance indicators, measurements) for evaluating safety culture?

Answer 6 [response from Sonja Haber, Human Performance Analysis Corporation]: The IAEA has a framework for the characteristics important for a positive safety culture. Each of the 5 characteristics has attributes that are used to evaluate the absence or presence of the characteristic. Those attributes are used in assessing the safety culture of an organization.

Question 7 [addressed to Sonja Haber, Human Performance Analysis Corporation]: How can we provide safety culture behaviors in NPPs if the regulators do not yet have internal safety culture programs? **This is from Dr. Haber's perspective in working internationally.**

Answer 7 [response from Sonja Haber, Human Performance Analysis Corporation]: There are several frameworks that define the characteristics, traits, and behaviors that are necessary for a healthy safety culture. These aspects of safety culture apply to the operating organizations as well as the regulatory bodies. The international nuclear community has adopted these frameworks and so every nuclear facility should be striving to achieve these elements for a healthy safety culture.

Question 8 [addressed to Sonja Haber, Human Performance Analysis Corporation]: You made the statement that probabilistic safety assessment assumes that nuclear power is safe. What is the basis for that statement? In practice, PSA requires analysts to think in "failure space", which is contrary to your statement.

Answer 8 [response from Sonja Haber, Human Performance Analysis Corporation]: My statement was predicated on the thinking of organizational theory that if we (as a nuclear community) operate and behave on the basic assumption that nuclear is safe, all of the thinking that goes into our work has this as a premise, whether it be conscious or unconscious. Consequently, without being aware of it, we may be designing our tools, such as PSA, using this basic assumption and not challenging ourselves enough to think outside that belief.

Question 9 [addressed to Andrew Lawrence, DOE]: What do you see as the most significant safety culture deficiencies identified by the Accident Investigation Board following the February 2014 incidents at the Waste Isolation Pilot Plant? How will DOE address them?

Answer 9 [response from Andrew Lawrence, DOE]: The most significant safety culture challenges identified by the Accident Investigation Boards (AIB) following the February 2014 incidents at the Waste Isolation Pilot Plant (WIPP) are the existence of several different kinds of cultures, across several different organizational levels, including Federal and contractor personnel.

The AIB identified, in these events, that there were contributing factors resulting from actions from not just at WIPP but at several DOE sites and organizations. DOE is not just addressing the multiple cultures at WIPP, such as the mining culture, and nuclear safety culture, but also the cultures at other sites and organizations that work with WIPP.

At this time it is too soon to state how the action plans will address the significant safety culture challenges. There are still other reports being finalized and additional information is flowing between and among headquarters; the sites and DOE site offices; and DOE Federal and contractor organizations. All of these organizations are directly involved with the evaluation and analysis of each AI Report's Judgments of Needs, and development of the implementation plans for corrective actions. As part of this process, DOE will address organizational, system, and human performance improvement to prevent similar events and assure the safe mission performance of the DOE National Transuranic (TRU) Waste Program.

Question 10 [addressed to Andrew Lawrence, DOE]: Why did the safety culture measured at WIPP before its incident change so much to that measured after the events? If a focus on safety culture cannot prevent such events, why even waste the time and effort?

Answer 10 [response from Andrew Lawrence, DOE]: Measuring safety culture at the Department of Energy is a new journey. It should be noted that safety culture change is evolutionary, taking years before improvements can be measured. There is no single instrument used to date to offer a comparable safety culture study of WIPP from past to present, so it is not possible to document any change in WIPP's safety culture at this point. We continue to reach out to experienced and successful organizations to learn from their best practices. The independent safety culture assessment of WIPP performed by the Institute for Nuclear Power Operations is in the final stages of document review. It will provide another set of objective observations for NWP to base its improvement actions.

Question 11 [addressed to Andrew Lawrence, DOE]: If each of you brought in an employee of your organization and asked them about safety culture, what would they say? (Please don't use the party line).

Answer 11 [response from Andrew Lawrence, DOE]: He or she would say: "I believe that I work in an environment where my safety is of paramount concern to my management. It is a place where I understand my job and its hazards and am free to voice any concerns I have with regard to either knowing that I will be listened to with respect and receive a fair Answer to my Questions. Safety culture is something I can both 'see' in the efforts my management takes to keep me safe and 'feel' in the sense of shared ownership for safety that I have with my supervisors and my fellow workers."

Question 12 [addressed to Andrew Lawrence, DOE]: Has DOE benchmarked/leveraged/adopted "best practices" and "methods" from the Naval Reactors Programs which is considered by many to be a role model for high performance organizations with an excellent record of "nuclear safety"?

Answer 12 [response from Andrew Lawrence, DOE]: DOE has looked at, and continues to look at, the best practices and methods from a variety of organizations that have devoted efforts to improving their safety cultures. While there are organizations from the Naval Reactors Program to those in the private sector who have maintained excellent safety records while performing high-hazard operations, it is difficult to "import" the safety culture from one organization to another. No two cultures are exactly alike, and what works well in one setting may not be suitable elsewhere. One of the reasons DOE developed its own definition of safety culture, rather than adopting one from another organization, was for it to be specific to its environment and its people and create a shared sense of "ownership."

Question 13 [addressed to Andrew Lawrence, DOE]: Because most of DOE's work is done by contractors, how does the presentation apply to DOE contractors? Do DOE contractors use DOE programs or do the contractors have their own programs?

Answer 13 [response from Andrew Lawrence, DOE]: Contractors have been an integral part of DOE's safety culture improvement efforts. They were deeply involved in DOE's response to Defense Nuclear Safety Board Recommendation 2011-1, Safety Culture at the Waste Treatment and Immobilization Plant, and played a lead role in developing DOE's safety culture definition currently included as part of the Department's directives system. Through organizations such as Energy Facility Contractors Group, EFCOG, we practice collaboration in our pursuit for excellence and not just compliance. DOE's safety culture goals flow down to its contractors through contractual clauses and inclusion of DOE's Integrated Safety Management System requirements into the Department of Energy Acquisition Regulations which apply to all of our contractors.

Question 14 [addressed to Andrew Lawrence, DOE]: Some believe the focus on SC is excessive or misfocused. Why does DOE believe it's an appropriate use of resources to devote to SC?

Answer 14 [response from Andrew Lawrence, DOE]: There exist a number of validated tools by which safety culture can be assessed on a regular basis without creating an excessive burden on either management or the workforce. As I said in my presentation, you cannot have good safety performance without a good safety culture. While there are many regulations governing occupational safety on the books, unless a culture exists where workers feel ownership of their own safety and the ability to identify, without fear of retribution, Questions or concerns about the job hazards that they face, valid safety issues can and will be driven underground. As we have seen from our own accident investigation program, a flawed safety culture is a prescription for serious consequences for worker safety and health. In such cases, an ounce of prevention is worth much more than a pound of cure.

Question 15 [addressed to Lori Hayes, Duke Energy]: It looks like you have a systematic process to monitor your safety culture. If your oversight process involves people at your company, how do you avoid the trap of not seeing gaps that the team has grown accustomed to?

Answer 15 [response from Lori Hayes, Duke Energy]: The Nuclear Oversight organization is a corporate led function. Challenge calls as well as leadership challenges from corporate for each site is embedded into the procedure, to ensure independence remains. In addition, audit and performance assessment teams are comprised of Team Leads and members from different sites other than the site being reviewed. This provides “fresh eyes” on culture and implementation of process. In addition, Duke’s Employee Concerns Program reports up through the Nuclear Oversight Organization Vice President as well, and provides additional independence.

Question 16 [addressed to Lori Hayes, Duke Energy]: What are the biggest challenges you see that the licensees have in meeting NRC expectations for safety culture from your perspective? What can you do to fix or meet these challenges?

Answer 16 [response from Lori Hayes, Duke Energy]: One of the biggest challenges is new supervisors or leaders who do not understand how their actions or inactions affect nuclear safety culture and the trust of those who they are leading. The challenge to get all new supervisors trained in this behavior recognition before they begin their role is difficult and we continue to see some negative effects from this. In addition, longstanding leadership shortfalls that are not addressed in a timely manner often send a message to individual contributors that leadership is “untouchable”. What may start out as one instance of less than stellar behavior not being addressed, may become an eroding nuclear safety culture.

Question 17 [addressed to Lori Hayes, Duke Energy]: As you merged the two utilities and Duke Power, what successful things did you do for successfully executing change management so that all leaders understand and support a healthy safety culture?

Answer 17 [response from Lori Hayes, Duke Energy]: Early on in the pre-integration design work, we recognized the need for a strong change management program to help employees navigate the many changes planned as we work towards “One Team, One Fleet, One Company”. A proven change management methodology was selected after benchmarking others in the nuclear industry and in the academia arena. A change management procedure was created and now governs our change management process for any significant changes impacting our employees, integration or otherwise. Our procedure establishes the process for identifying, evaluating, developing, executing, and monitoring the effectiveness of change management at the fleet, corporate, or individual site level. The desired outcome is to manage changes in a consistent manner in order to reduce human errors and increase the effectiveness of change implementation.

Question 18 [addressed to Lori Hayes, Duke Energy]: Can you share some successes or best practices to strengthen employee engagement throughout the organization--vertically and horizontally?

Answer 18 [response from Lori Hayes, Duke Energy]: A trend identified at one station's Nuclear Safety Culture Monitoring Panel was Respectful Work Environment. Upon review of that weakness, the Duke fleet subsequently determined this was a weakness at some of our other stations as well. One of the actions taken was to reconstitute our employee engagement efforts. We now have employee engagement teams at our stations in which employees identify areas for improvement with a facilitator without management presence. Management then addresses the employee engagement team with the Answer. It may not be the Answer some employees are happy about, but the Answer is provided. One of the areas we determined that needed to be strengthened was feedback to employees. With the high volume of changes occurring, some employees did not feel they were being heard.

Question 19 [addressed to Lori Hayes, Duke Energy]: The NRC has revised their safety culture policy statement to include vendors of safety related components. Has Duke begun to assess their supplier's safety culture? Is it being added to the GOSP program?

Answer 19 [response from Lori Hayes, Duke Energy]: Vendors and suppliers have always been included in the NRC's safety culture policy statement. For maintenance alliance, engineering of choice, or other large contracts, Duke Energy performs a surveillance of those vendors' procedures and processes regarding nuclear safety culture and Safety Conscious Work Environment. This is typically completed by Employee Concerns after the contract is awarded. This is part of the GOSP program, in that corporate establishes, reviews and performs these functions.

Question 20 [addressed to Lori Hayes, Duke Energy]: Can utilities/NRC distinguish between a hostile work environment of harassment and bullying (which NRC has little authority over) and an environment in which people fear raising safety issues (SCWE which NRC does have some leverage over)?

Answer 20 [response from Lori Hayes, Duke Energy]: There is a close link to both. If the hostile work environment is determined to be an adverse action due to an individual raising a safety concern, at Duke we engage our legal organization to assist in determining if protected activity is involved. If an adverse action, such as hostile work environment is substantiated to have occurred, as part of that investigation, Duke Employee Concerns would interview or survey individuals to determine if there is any fear of raising concerns.

Question 21 [addressed to Lori Hayes, Duke Energy]: Please describe a trend identified by your monitoring program and the action taken to address it.

Answer 21 [response from Lori Hayes, Duke Energy]: A trend identified at one station's Nuclear Safety Culture Monitoring Panel was Respectful Work Environment. Please see Answer to Question 18.

Question 22 [addressed to Lori Hayes, Duke Energy]: If each of you brought in an employee of your organization and asked them about safety culture, what would they say? (Please don't use the party line).

Answer 22 [response from Lori Hayes, Duke Energy]: For the most part, I believe individuals would say Duke has a good safety culture. We have heard from individuals they miss "the good old days" and "we used to be a family." However, our stations are now part of a larger fleet, many individuals have moved to other stations, and accountability and leader behavior strengthening continues to be emphasized.

TECHNICAL SESSIONS
Wednesday, March 11, 2015, 3:30 p.m. – 5:00 p.m.

W22 Design Integrity throughout the Supply Chain

Session Chair: Richard Rasmussen, Branch Chief, Division of Construction Inspection and Operational Programs, NRO/NRC, 301-415-1340, Richard.Rasmussen@nrc.gov

Session Coordinator: Michelle Hayes, Technical Assistant, Division of Construction Inspection and Operational Programs, NRO/NRC, 301-415-8375, Michelle.HayesNRO@nrc.gov

The questions below were not answered during the above session.

Question 1 [addressed to all]: Today's supply chain for Appendix B, Nuclear Safety Related Items, inherently involves commercial grade items from sub-tier suppliers. What are some best practices for design verification of third party sub-tier suppliers?

Answer 1 [response from Joselito Calle, TVA]: TVA has the following procurement note that is invoked on Appendix B suppliers:

"Commercial grade material procured from a source not qualified in accordance with the supplier's quality program and dedicated by the supplier, to be furnished to TVA as safety-related, must have all critical attributes (e.g., chemistry, tensile, hardness, etc.) required by the applicable material specification and identified on the manufacturer's material test report independently verified. Independent verification must be performed by the supplier or a laboratory qualified by the supplier in accordance with his safety-related quality program requirements and documented dedication practices."

The specific commercial grade dedication plan used by the Appendix B supplier is submitted to the TVA site Procurement Engineering Group who issued the purchase order for their review before the material is dedicated. The applicable CGD package must also be provided as part of the procurement documentation that accompanies the dedicated material and is reviewed as part of the receipt inspection process.

In addition, an audit of the Appendix B supplier's commercial grade dedication program is performed before the supplier is placed on the Approved Supplier's List as applicable. In any case, the supplier's CGD program is audited before it is used.

W23 Emergency Preparedness Applied Research

Session Chair: Robert Kahler, Branch Chief, Division of Preparedness and Response, NSIR/NRC, 301-287-3756, Robert.Kahler@nrc.gov

Session Coordinator: Edward Robinson, Emergency Preparedness Specialist, Division of Preparedness and Response, NSIR/NRC, 301-287-3774, Edward.Robinson@nrc.gov

The questions below were not answered during the above session.

Question 1: How is Emergency Preparedness & Hostile Action Exercises incorporating the possibility of a cyber-attack in combination with a terrorist attack?

Answer 1: Hostile Action, as defined in Appendix E to 10 CFR Part 50, is an act directed toward a nuclear power plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Cyber security regulatory requirements are located in 10 CFR 73.54, "Protection of digital computer and communication systems and networks" and a cyberattack is not considered to be hostile action. As such, cyberattacks are not incorporated in HAB exercises. The HAB exercise requirement is an evaluation of the emergency plan implementation during a hostile action event and the support/coordination provided by the site's physical security plan. The HAB exercise requirement in Appendix E to 10 CFR 50 uses a security initiated event to establish conditions that challenge the emergency response organization's (ERO) ability to implement the emergency plan and take mitigative actions. Further, an objective of the HAB exercise is to demonstrate the coordination and communications between the license's Operations, Security and ERO with the local emergency management agency and first responders (e.g., LLEA, tactical law enforcement, firefighting, and emergency medical assistance).

Question 2: One of the lessons of Fukushima is that the EPA PAGs for evacuation were likely exceeded at approximately 25-miles, and potentially could have been exceeded as far away as Tokyo (data from DOE and NRC are available to support this). Why isn't this sufficient?

Answer 2: The EPA PAGs for evacuation were not exceeded at 25 miles, however the PAGs for relocation were calculated to be exceeded and that is why that population was relocated. Evacuation is the urgent removal of people from an area to avoid or reduce high-level, short-term exposure, from the plume or from deposited activity. Whereas, relocation is the removal or continued exclusion of people (households) from contaminated areas to avoid chronic radiation exposure. The NRC continues to monitor domestic and international efforts and events for potential enhancements to the NRC's EP regulatory framework and guidance. The NRC remains confident that the existing emergency planning zones, framework and regulations provide reasonable assurance of adequate protection of public health and safety in the event of a radiological emergency at a U.S. power reactor. NRC staff closely follows the ongoing Fukushima health studies conducted by international organizations as well as by the Japanese authorities. Initial review of these studies does not appear to challenge the EP planning basis.

W24 International Approaches to Low-Level Radioactive Waste Management—Key Issues and Challenges

Session Chair: Larry Camper, Director, Division of Decommissioning, Uranium Recovery, and Waste Programs, NMSS/NRC, 301-415-6673, Larry.Camper@nrc.gov

Session Coordinator: Gregory Suber, Branch Chief, Division of Decommissioning, Uranium Recovery, and Waste Programs, NMSS/NRC, 301-415-8087, Gregory.Suber@nrc.gov

The questions below were not answered during the above session.

Question 1 [addressed to Boby Abu Eid, NRC]: With the changes in NRC regulation of accelerator produced material as byproduct material (now licensed), how have NRC regulations for disposal changed?

Answer 1 [response from Boby Abu Eid, NRC]: Accelerator produced material is treated as byproduct material in NRC's regulations (see byproduct material definition in 10 CFR 30.4) and would be disposed of in a manner similar to other Atomic Energy Act Section 11e materials. In general, the NRC's direct responsibility for disposal of such materials can be transferred to the Agreement States and NRC would conduct periodic reviews of such material disposal under the Integrated Materials Performance Evaluation Program.

In brief, we do not believe that NRC regulations under 10 CFR Part 61 need to be changed to account for accelerator produced material as such byproduct materials are already regulated under 10 CFR Part 30 or by the Agreement States' compatible regulation.

Question 2 [addressed to Boby Abu Eid, NRC]: Based on NRC experience, are estimated quantities of GTCC from operating nuclear power plants (NPPs) or decommissioning NPPs close to practical quantities?

Answer 2 [response from Boby Abu Eid, NRC]: NRC regulations under 10 CFR Part 61 involves four categories of low-level waste; namely Classes A, B, C, and Greater-than-Class-C (e.g.; GTCC).

GTCC waste sources can be divided into three main categories: (a) **Activated Metals** (largely generated from the decommissioning of nuclear reactors); (b). **Sealed Sources** (largely from industrial, medical, and academic uses); and (c) **Other Waste** (mostly related to production of isotopes).

NUREG-1713, October 2004, provided the following estimates of volumes of LLW waste categories resulting from DECON of NPPs:

For **BWR:**
Class A: 514,723 ft^3, 14,575.3 m^3 (96.37%)
Class B/C: 19,152 ft^3, 542.3 m^3 (3.59%)
GTCC: 244 ft^3, 6.9 m^3 (0 .05%)

For **PWR:**
Class A: 280,900 ft^3, 7,954 m^3 (96.5%)
Class B/C: 9,000 ft^3, 255 m^3 (3.4%)
GTCC: 400 ft^3, 11.3 m^3 (0 .13%)
It is noted that NRC staff is currently reviewing the calculations for LLW produced during NPPs decommissioning activities.

The US DOE issued a draft environmental impact statement (EIS) Published in February 2011 (DOE/EIS-0375-D) regarding Disposal of GTCC waste. The draft EIS listed in Table S-1 the following data regarding GTCC inventory:

Waste Type	In Storage		Projected		Total Stored and Projected	
	Volume (m^3)	Activity (MCi)[b]	Volume (m^3)	Activity (MCi)	Volume (m^3)	Activity (MCi)
Group 1						
GTCC LLRW						
Activated metals (BWRs)[c] – RH	7.1	0.22	200	30	210	31
Activated metals (PWRs) – RH	51	1.1	620	76	670	77
Sealed sources (Small)[d] – CH	–[e,f]	–	1,800	0.28	1,800	0.28
Sealed sources (Cs-137 irradiators) - CH	–	–	1,000	1.7	1,000	1.7
Other Waste[g] – CH	42	0.000011	–	–	42	0.000011
Other Waste - RH	33	0.0042	1.0	0.00013	34	0.0043
Total	130	1.4	3,700	110	3,800	110

We believe the projected total volume from BWRs and PWRs by 2035 could be higher due to an increase in number of commercial reactors shutdown and subsequent increase in power reactor decommissioning activities.

W25 Update Process for Approved Transient and Accident Analysis Methods

Session Chair: Jeremy Dean, Branch Chief, Division of Safety Systems, NRR/NRC, 301-415-1008, Jeremy.Dean@nrc.gov

Session Coordinator: Kevin Heller, Reactor Systems Engineer, Division of Safety Systems, NRR/NRC, 301-415-8379, Kevin.Heller@nrc.gov

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

W26 Regulatory Agility in the New Millennium

Session Chair: Michael F. Weber, Deputy Executive Director for Operations for Materials, Waste, Research, State, Tribal, and Compliance Programs, OEDO/NRC, 301-415-1705, Michael.Weber@nrc.gov

Session Coordinator: Cindy Rosales-Cooper, Executive Technical Assistant, OEDO/NRC, 301-415-1168, Cindy.Rosales-Cooper@nrc.gov

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

TECHNICAL SESSIONS **Thursday, March 12, 2015, 8:30 a.m. – 10:00 a.m.**

TH27 Defense-in-Depth: A Historical Perspective within a Dynamic Regulatory Framework

Session Chair: Gary Holahan, Deputy Director, Office of New Reactors, NRO/NRC, 301-415-1897, Gary.Holahan@nrc.gov

Session Coordinator: John Nakoski, Branch Chief, Division of Risk Analysis, RES/NRC, 301-251-7612, John.Nakoski@nrc.gov

The questions below were not answered during the above session.

Question 1 [addressed to NRC OR NEI]: Defense in depth is always cited as a qualitative, non-quantifiable concept. Yet there have been efforts within NRC itself (& maybe industry?) to align it with safety margin on a quantifiable basis usually through probabilistic, statistical means. Is there any effort planned to pursue these preliminary steps so that defense in depth (& safety margin) is better integrated into risk informed regulation than the current qualitative-only link through RG 1.174?

Answer 1 [response from Mary Drouin and John Nakoski, NRC]: At the moment, there is not concerted effort to quantify the application of defense-in-depth into NRC's regulatory decision-making processes. Recognizing however, that the NRC is looking at how risk management is implemented, it is natural to conclude that consideration of enhancing the integration of defense-in-depth into its risk-informed decision-making process will occur. Also, the NRC recognizes that PRA insights can provide input to where there may be either inadequate or adequate defense-in-depth. However, this would just be one factor in determining the adequacy of defense-in-depth. Other factors would also be included in the decision criteria for evaluating the adequacy of defense-in-depth. There are a few examples where Defense-in-Depth is quantified. The Safety Goal subsidiary goals (1E-04/yr core damage frequency (CDF) and 1E-05/yr LERF (larger early release frequency for operating reactors) represent a type of DiD quantification. A similar concept is included in the screening guidance for backfits (NUREG /BR-0058 Figure 3.2). In addition, the risk-informed regulatory framework suggested for next generation plants included a similar "risk allocation" concept (NUREG-1860).

Question 2 [addressed to NRC]: Could one of the NRC panelist comment on the notion of flexibility raised by Mr. Reig?

Answer 2 [response from John Nakoski, NRC]: Flexibility in the application of defense-in-depth is consistent with the NRC's approach to assuring that the preventive and mitigative measures used to assure public health and safety are consistent with the potential consequences from the specific regulated activity. Further, the application of defense-in-depth needs to be sufficiently flexible, as it relates to nuclear power plants in particular, to account for unique design features, operational practices, and site characteristics to assure the preventive and mitigative measures are sufficient to assure the risks are adequately managed. The fact that DiD is a philosophy or a concept rather than a set of rules means that it is inherently flexible.

Question 3 [addressed to Mary Drouin, NRC]: You mentioned that it's difficult to understand how to implement DID but didn't discuss cases where plants might be unwilling to implement adequate DID for unknowns. For instance, App. R has pretty clear requirements yet no plant complies with them w/o exemptions.

Answer 3 [response from Mary Drouin, NRC]: Implementing the DID philosophy is difficult and often requires consideration of factors that are specific to a plant design, site conditions, and the licensee's approach to assuring the safe operation of the facility. Using the example you provided related to Appendix R and granting of exemptions to requirements, the NRC follows a well-established process for reviewing the licensee's request for the an exemption. This process includes consideration of alternative methods for achieving the underlying purpose of the requirement – in essence the NRC makes a determination that there are adequate alternative methods (defensive barriers) to compensate for the licensee not complying with a requirement in an NRC regulation. In doing so, the NRC makes a decision that there is adequate defense in depth.

Question 4 [addressed to Jennifer Uhle, NRC]: The presentations illustrate the challenges with applying the DID philosophy. Don't all those challenges also exist with the basic mandate of "adequate protection"? Can you fully resolve the DID challenges without further addressing "adequate protection"? The Commission's White Papers don't fully define adequate protection. Should DID be fully separated from consideration of adequate protection?

Answer 4 [response from John Nakoski, NRC]: Defense-in-depth and "adequate protection" is complementary concepts that are used in regulatory decision-making. And, yes many of the same challenges exist when applying these concepts. However, steps can be made to better define the appropriate application of DID, without necessarily further addressing "adequate protection." As the NRC progresses in its thinking on how to better measure the effectiveness of DID, it will likely also inform its understanding of what "adequate protection" means and how DID relates to this understanding.

Question 5 [addressed to Mary Drouin, NRC]: You stated on Tuesday that PRA uncertainty analyses per NUREG-1855 do not account for the U/U [unknown unknowns] unknowns, and that D in D is required for such uncertainties. How is use of PRA results to determine adequacy of D in D not a circular argument?

Answer 5 [response from Mary Drouin and John Nakoski, NRC]: A PRA model give insights with regard to the strength and weakness of the design and operation of the plant, and can provide insights regarding how well the quantitative acceptance guidelines are met. This is not a circular argument. However, how well quantitative acceptance guidelines are met is just one factor in determining the adequacy of defense-in-depth; it does not provide insights regarding the unknown unknowns, this determination would be made by other decision criteria, for example, evaluating safety margins, how will layers of defense-in-depth have been implemented.

Question 6 [addressed to all]: Could you give us a clearer idea of when the draft NUREG on defense in depth will be released for comment and how we can stay apprised of this so we can comment?

Answer 6 [response from Mary Drouin, NRC]: The draft NUREG should be available for public comment in the fall of 2015.

Question 7 [addressed to Mary Drouin, NRC]: Are you considering prevention and mitigation as two barriers towards the Defense-in-Depth principle, or multiple barriers within prevention and mitigation as Defense-in-Depth?

Answer 7 [response from Mary Drouin, NRC]: As a major principle, defense-in-depth must provide for both prevention and mitigation protective measures. There are two ways to look at prevention and mitigation. At a high level, you want both prevention and mitigation of an accident. At a lower level, you have prevention and mitigation in multiple places; for example,

- Prevent an adverse event from occurring
- Mitigate the consequence of the adverse event
- Prevent the adverse event progressing to a core damage state
- Mitigate the consequence of a core damage state
- Prevent core damage progressing to a release
- Mitigate the consequence of a release
- Etc.

Question 8 [addressed to Javier Reig, NEA]: Why USNRC/US Industry and NEA have different challenges? If different, why not make a common list of challenges?

Answer 8 [response from John Nakoski, NRC]: The NRC, the US Industry, and the NEA Member states face the same challenges in implementing the concept of defense-in-depth. While there may be subtle differences in how the challenges are described and addressed, the challenges are essentially the same.

Question 9 [addressed to all]: Defense in depth by adding multiple barriers compensates for uncertainties and unknown unknowns. But the additional barriers and layers also increase the complexity of the system – and thus the possibility of human failure and unanticipated technological failure. How does one think about balancing these two things?

Answer 9 [response from Mary Drouin and John Nakoski, NRC]: Part of the problem in discussing defense-in-depth is the terminology. Barriers in some cases are only meant to mean physical barriers, while in other cases it is meant to mean a protective measure (design, operational or programmatic feature) meant to prevent or mitigate. Nonetheless, these barriers or layers are not something that is added on after the design, etc. is decided but it is integrated as part of and, inherent to the design, etc.

Question 10 [addressed to Jennifer Uhle, NRC]: Licensees frequently submit license amendment requests that propose a change (similar to your examples) without following RG 1.174, yet they provide risk information to support the basis for their request. While insightful, the current regulatory framework does not address these situations or provide a reasonable means for traditional deterministic technical review branches to deal with such risk information in an appropriate and consistent manner.

Until the regulatory framework is further enhanced and guidance developed, how should the NRC staff treat any risk information provided by an applicant/licensee that doesn't follow RG 1.174?

Answer 10 [response from John Nakoski, NRC]: The NRC has internal procedures that govern the review of license amendment requests (such as, LIC-101, "License Amendment Review Procedures;" LIC-501, "Program Coordination for Risk-Informed Activities, etc.) that NRC staff follows in the review of license amendment requests. While these internal procedures do not cover every circumstance, they do provide sufficient guidance for most of the licensing reviews. When a license amendment request includes risk information but does not follow the guidance provided by RG 1.174, the staff will include the information within the scope of its review, although it may not have a significant impact on the outcome of the staff's decision-making process since it would not likely provide the staff with sufficient information to make a risk-informed decision. In these instances, the burden is on the licensee to assure that the information it provided is sufficient to support its request based on the traditional, deterministic approach, to staff reviews.

TH28 Perspectives on the New Reactor Licensing Process

Session Chair: Mark Delligatti, Deputy Director, Division of New Reactor Licensing, NRO/NRC, 301-415-1199, Mark.Delligatti@nrc.gov

Session Co-Coordinators: Alexandra Burja, NSPDP General Engineer, Division of New Reactor Licensing, NRO/NRC, 301-415-6144, Alexandra.Burja@nrc.gov

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The questions below were not answered during the above session.

Question 1 [addressed to NRC]: Discuss the relationship between NRC and Environmental Protection Agency (EPA) reviews. Are they separate? Coordinated?

Answer 1 [response from Alicia Williamson-Dickerson, NRC]: Section 309 of the Clean Air Act gives the U.S. EPA the responsibility to review environmental impact statements (EISs) that are prepared by other Federal agencies, including the NRC. In addition to reviewing EISs for adequacy, the EPA also provides the sponsoring agency (in this case, the NRC) with an assessment of each draft EIS as a measure of the NRC's adherence to the National Environmental Policy Act (NEPA) using the EIS rating system criteria (http://www.epa.gov/compliance/nepa/comments/ratings.html). These rating criteria provide a basis upon which EPA makes recommendations to the NRC for improving the draft EIS. In addition, the comments provide additional feedback to the NRC to use in developing future EISs. The NRC also invites other Federal agencies, including the U.S. Fish and Wildlife Service and National Marine Fisheries Service, to participate in the environmental review process and review and comment on the draft EISs.

Question 2 [addressed to Samuel S. Lee, NRC]: Can we have your idea for efficient request for additional information (RAI) interaction from both the staff side and applicant side?

Answer 2 [response from Samuel S. Lee, NRC]: As one of NRC's five Principles of Good Regulation, efficiency is highly encouraged. The NRC staff utilizes the following guidelines to achieve better efficiency with RAI interaction:

- Staff should issue RAIs only when information is needed to complete the technical review and there is missing or misleading information in the applications, i.e., to make a regulatory finding.
- RAIs should include clear and concise statement(s) of information needed and have an understandable regulatory basis.

Additionally, RAI interaction efficiency is improved when the applicant provides timely RAI responses that address the queries to their full extent and both the applicant and the NRC staff commit to communicating clearly throughout the RAI process.

Question 3 [addressed to Samuel S. Lee, NRC]: In order to avoid subsequent RAIs, clarification of the RAI is very important. Typically, the conference call is a preferred method to clarify RAIs. However, sometimes the applicant wants to have a face-to-face meeting to clarify RAIs. Is that possible or recommended by the NRC staff?

Answer 3 [response from Samuel S. Lee, NRC]: Face-to-face public meetings can be held to clarify RAIs; however, we believe the better use of such meetings is to resolve or reach closure on technical or regulatory matters that require extensive discussion. Conference calls are often used because they are more efficient in terms of time, money, and schedule for all parties involved.

Question 4 [addressed to Samuel S. Lee, NRC]: How are you using the lessons learned on the APR1400 pre-application activities for the NuScale pre-application activities? Is NuScale taking advantage of these lessons learned?

Answer 4 [response from Samuel S. Lee, NRC]: The NRC staff is applying the updated process for conducting pre-application activities with all future applicants. New applicants are made aware that, in order to better understand whether the applicant's design information will be sufficiently complete to support the NRC's rigorous acceptance review and a timely decision whether to docket the application, the NRC staff needs to preview design details during the pre-application stage. This expectation is made clear in the Part 52 Lessons Learned Report and the Office of New Reactors office procedures on (1) pre-application activities and (2) how to conduct acceptance reviews. The NRC staff is currently engaged in pre-application activities with NuScale.

Question 5 [addressed to Samuel S. Lee, NRC]: You mentioned that the APR1400 did not clear some design issues. One of them is digital instrumentation and control (I&C). Could you elaborate in detail the I&C design concerns?

Answer 5 [response from Samuel S. Lee, NRC]: Two main examples of digital I&C issues not sufficiently addressed by the applicant in its 2013 design certification application are (1) software common cause failures of non-safety related control systems that can lead to spurious actuations of redundant safety and non-safety components that could potentially exceed the plant safety analysis; and (2) critical characteristics, such as deterministic performance and the software development process for its safety-related digital I&C system platform.

Question 6 [addressed to Samuel S. Lee, NRC]: How much impact did the lack of a combined license (COL) applicant or U.S. construction have on the pre-application to the APR1400 Design Certification (DC) application? Does the U.S. NRC anticipate it will routinely certify non-U.S. origin designs without COL or construction targets in the U.S.?

Answer 6 [response from Samuel S. Lee, NRC]: First, the lack of a COL (or COL applicant) referencing a design does not preclude the development of a DC application. The value of a COL applicant is in providing further insights into the constructability of the design. Also, since KHNP has experience with constructing the design in both Korea and now the UAE, the lack of a COL application in the U.S. should not hinder or delay the review process for the DC. Per NRC Commission's approval, the staff engaged the applicant for the APR1400 design in pre-application space. The Commission has not decided whether the NRC will routinely engage in reviewing non-U.S. origin designs.

Question 7 [addressed to William Maher, Florida Power and Light]: What are the big books you brought with you?

Answer 7 [response from William Maher, Florida Power and Light]: The Turkey Point 6&7 Draft Environmental Impact Statement (EIS).

Question 8 [addressed to Mark Delligatti, NRC]: What is the NRC doing to get in front of design certification (DC) renewal applications from industry to provide guidance to make the process efficient for all involved?

Answer 8 [response from Mark Delligatti, NRC]: The NRC initiated a lessons learned review to identify potential enhancements to the Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," licensing process and contribute to more effective and efficient reviews of future applications. To facilitate this lessons learned review, the NRC conducted an outreach effort to solicit feedback from external and internal stakeholders on their experiences using the new reactor licensing process. Specifically, the NRC staff drew on previous assessments of portions of the new reactor licensing process, lessons shared at the NRC's 2012 Regulatory Information Conference, feedback received at a public meeting on lessons learned, and the results of internal and external surveys on the new reactor licensing process.

The NRC and its stakeholders have committed to engage in open and transparent communications, in a timely manner, which will continue to result in the successful implementation of the Part 52 licensing process. In addition, the NRC identified several planned and potential actions that can be used to enhance the licensing process and improve the efficiency of future reviews of design certifications (and can be applied to design certification renewal applications), including but not limited to: the applicant's submittal of a complete and high-quality application; the NRC staff continuing its commitment to a continuous, ongoing effort to update guidance; early identification and resolution of technical issues; and resolving all design issues before rulemaking begins.

More specific details can be found in NRC's report, "New Reactor Licensing Process Lessons Learned Review: 10 CFR Part 52," (Agencywide Documents Access and Management System Accession No. ML13059A239).

Question 9 [addressed to Mark Delligatti, NRC]: The December 2011 Federal Register notice describes the NRC consideration to adopt “Branches” alternatives as the regulatory approach for multiple suppliers of the same design certification for renewal. If a DC renewal application submitted by Supplier “A” is approved while the other DC renewal application submitted by Supplier “B” (for the original design certification document [DCD]) is under NRC review, is the original DCD still effective, and can it be referenced by the existing or future combined license (COL) applicants?

Answer 9 [response from Mark Delligatti, NRC]: In accordance with 10 CFR 52.55(b) and 52.57(b), a standard design certification remains in effect beyond its expiration date and can continue to be referenced by a COL applicant if (1) a timely application to renew the design certification has been submitted and (2) a COL application referencing the originally certified design is docketed before the Commission has determined whether to renew the certification. NRC regulations, however, do not directly address a prospective COL applicant’s ability to reference the original DCD under various hypothetical scenarios associated with multiple applications to renew a design certification, nor is there any established NRC guidance or policy on this issue. Also, the U.S. Advanced Boiling Water Reactor is the only design certification that is currently the subject of a renewal application, and the NRC staff is not aware of any prospective COL applicant that wishes to reference the original DCD as opposed to the renewal DCDs currently under review. Therefore, the staff will not at this time address the hypothetical question raised by the asker. If the NRC determines a need to address this question in the future, it will do so through an appropriate vehicle.

Question 10 [addressed to Mark Delligatti, NRC]: If a DCD is being renewed and a subsequent combined license application (S-COLA) is submitted to the NRC before or after the NRC certifies the renewed DCD, can the S-COLA applicant use/refer to the original DCD that the reference combined license application (R-COLA) used?

Answer 10 [response from Mark Delligatti, NRC]: As discussed in the answer to Question 9, a prospective applicant’s ability to reference an original DCD is dependent upon whether the Commission has made a determination on whether to renew the DC. Thus, in the straightforward case where the original vendor applies for renewal in a timely manner and no other renewal applications exist, an S-COLA could reference the original DCD if the S-COLA is docketed prior to the NRC making its determination on the renewal application. In such a case, the staff could theoretically use its review of the initial R-COLA to support the S-COLA review to the extent that the S-COLA does not depart from the R-COLA approach. However, if the staff has completed its review of the renewal application and has incorporated the renewal DCD by reference into the design certification rule, then an S-COLA docketed after this point could not reference the original DCD because it has been superseded by the renewed DCD.

TH29 Reactor Decommissioning Transition, 1 Year Later

Session Chair: Larry Camper, Director, Division of Decommissioning, Uranium Recovery, and Waste Programs, NMSS/NRC, 301-415-6673, Larry.Camper@nrc.gov

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

The questions below were not answered during the above session.

Question 1 [addressed to Bob Orlikowski, NRC]: For planning purposes, what is the NRC view on unrestricted release standards for reactors in states that have more restrictive release standards? Will the NRC insect to the lower standard?

Answer 1 [response from Bob Orlikowski, NRC]: The NRC will inspect to the Federal release standards unless the reactor license specifically lists the more restrictive standard.

Question 2 [addressed to Bob Orlikowski, NRC]: Has 10 CFR Part 37 introduced any new issues in decommissioning plant inspections?

Answer 2 [answered to Bob Orlikowski, NRC]: Not yet. Many sites that are undergoing decommissioning have an approved security plan that meets the requirements of 10 CFR Part 73. Typically, these sites will maintain their security plan until they no longer possess category 1 or 2 material, or they reduce the boundary of their secure area to encompass only the category 1 or 2 material. As decommissioning activities progress at a site, the NRC will continue to evaluate how the licensee is meeting the 10 CFR Part 37 requirements.

Question 3 [addressed to Bob Orlikowski, NRC]: With regards to new technology, what does the NRC view as the most important aspect of looking at the effectiveness of new technology, and when is it most important?

Answer 3 [response from Bob Orlikowski, NRC]: The NRC performs inspections to ensure that the licensee is meeting the required regulations and the conditions of their license. The NRC may inspect activities that employ new technology, but the inspection focus will be how the new technology meets the requirements or license. The NRC does not perform inspections of new technology to evaluate anything other than safety or regulatory requirements such as cost or efficiency.

Question 4 [addressed to Douglas Broaddus, NRC]: Is the existing MOU between the NRC and FEMA being considered for changes to address the transition to decommissioning, perhaps along with the rulemaking efforts?

Answer 4 [response from Douglas Broaddus, NRC]: NRC rulemaking efforts that reduce offsite emergency planning requirements for decommissioning power reactors will need to be coordinated with FEMA. This may include changes to memorandum of understandings between FEMA and the NRC or may even involve conforming rulemaking by FEMA.

Question 5 [addressed to Douglas Broaddus, NRC]: Please describe some of the ways licensee are segregating spent fuel management costs from decommissioning fund and how these methods address circumstances where the funds were commingled in the past.

Answer 5 [response from Douglas Broaddus, NRC]: All four power reactor licensees that have recently had reactors permanently shut down, defuel, and enter into decommissioning have request exemptions to utilize excess monies in their respective decommissioning trust funds for irradiated fuel management. In the exemption requests, the licensees provide necessary cost data and fund growth estimates to demonstrate that the amount of the decommissioning trust fund that will be used for irradiated fuel management will not prevent the licensee from completing radiological decontamination of the decommissioning reactors. The staff will independently confirm that there is a reasonable assurance that the use of part of the decommissioning trust fund for irradiated fuel management will not impact the licensee's radiological decontamination of the site. In addition, the NRC staff reviews the overall status of the decommissioning trust fund yearly and will reassess that the remaining funds in the trust are adequate to complete decommissioning.

Question 6 [addressed to Douglas Broaddus, NRC]: Why do you need exemptions to use funds in waste management during cleanup after cease of operations under the PSDAR?

Answer 6 [response from Douglas Broaddus, NRC]: After a licensee has certified that it has permanently ceased operation and permanently defueled the reactor in accordance with Title 10 of the Code of Federal Regulations, Section (§) 50.82, up to 3% of the decommissioning trust fund can be used for decommissioning planning. Commencing 90 days after the Post Shutdown Decommissioning Activities Report (PSDAR) is submitted to the NRC staff for review, an additional 20% of the decommissioning trust fund may be used for legitimate decommissioning activities consistent with the definition of decommissioning in § 50.2. A site-specific decommissioning cost estimate must be submitted to the NRC prior to the licensee using any funding in excess of these amounts. The management of radioactive waste generated during site radiological decontamination does not require an exemption for use from the decommissioning trust fund since such actions are legitimate decommissioning activities. However, costs associated with the management of irradiated fuel are not considered radiological decommissioning and the licensee will have a separate funding plan developed under § 50.54(bb). The licensee must seek an exemption for § 50.82 and § 50.75 that demonstrates the use of part of the decommissioning trust for irradiated fuel management will not prevent completion of radiological decontamination of the site within 60 years. The NRC staff must independently confirm the licensee's assessment before consideration of granting such an exemption.

Question 7 [addressed to Douglas Broaddus, NRC]: What regulatory basis is being used to ask plants in decommissioning to maintain commitments made during license renewal, or to make new commitments in the area, with respect to aging management of passive components? If it is 50.51(b), where do we draw the line before it is a backfit?

Answer 7 [response from Douglas Broaddus, NRC]: For licensees that have received a renewed operating license, prior to the beginning of the period of extended operation, license renewal-related license conditions typically require that aging management programs and activities are incorporated in the updated final safety analysis report (UFSAR). Once these activities are incorporated in the UFSAR, modifications to the activities are evaluated using the requirements of 10 CFR 50.59. In accordance with 10 CFR 50.71(e)(6), the licensee is required to maintain the UFSAR throughout the operation and decommissioning of the plant, until the Commission terminates the license.

In accordance with 10 CFR 50.36(c)(6), facilities that have submitted the certifications required by 10 CFR 50.82(a)(1), are to develop technical specifications to reflect the decommissioning status on a case-by-case basis. During the NRC review of the license amendment request to modify a license and technical specifications to reflect the decommissioning status, the NRC evaluates the adequacy of safety limits, limiting safety system settings, and limiting control system settings; limiting conditions for operation; surveillance requirements; design features; and administrative controls. To the extent that an activity is necessary to maintain the facility in a safe condition, including, where applicable, the storage, control and maintenance of the spent fuel, the NRC would condition the license accordingly.

The regulatory basis for NRC action could consist of any of a number of regulatory provisions. In instances where a proposed NRC action would constitute backfitting, then the NRC would address the backfit requirements under 10 CFR 50.109.

Question 8 [addressed to Douglas Broaddus, NRC]: How is the post-Fukushima lessons learned applied to power reactors transitioning into decommissioning? What is the safety of spent fuel pools during transition?

Answer 8 [response from Douglas Broaddus, NRC]: The NRC staff has performed a preliminary assessment of the applicability of Fukushima lessons learned to facilities other than operating power reactors (Agencywide Documents Access Management System Accession No. ML15042A367). This preliminary assessment includes reactor transitioning to decommissioning and concludes that no action is required.

The safety of a spent fuel pool (SFP) during decommissioning transition is unchanged from that of an operating reactor. Changes to the regulatory framework involving SFPs during decommissioning must be submitted to the NRC for approval. The NRC staff would assess any such requests to ensure that the public health and safety is maintained. There are numerous studies that show SFPs are robustly designed structures that are likely to withstand severe earthquakes, such as Fukushima, without leaking. The SFPs will continue to provide protection of any irradiated fuel stored in the fuel throughout the decommissioning process.

Question 9 [addressed to Douglas Broaddus, NRC]: The proposed integrated rulemaking for beyond design basis events (Fukushima) addresses plants in transition to decommissioning to some extent, but may not be entirely aligned with recently approved licensing actions. How closely is this effort being coordinated with those involved in recent lessons learned on decommissioning transition?

Answer 9 [response from Douglas Broaddus, NRC]: The NRC staff has performed a preliminary assessment of the applicability of Fukushima lessons learned to facilities other than operating power reactors (Agencywide Documents Access Management System Accession No. ML15042A367). This preliminary assessment includes reactor transitioning to decommissioning and concludes that no action is required. The licensing actions approved for the recently permanently shut down power reactors are not inconsistent with Fukushima recommendations. As noted in the preliminary assessment, a Commission directed decommissioning rulemaking will provide an opportunity for both the NRC staff and public stakeholders to assess if Fukushima related rules are needed for decommissioning reactors.

Question 10 [addressed to Douglas Broaddus, NRC]: Legal source to establish decommissioning transition working group?

Answer 10 [response from Douglas Broaddus, NRC]: The NRC reactor decommissioning transition working group is an NRC internal organization to assess and evaluate reactor decommissioning issues. No special legal authority is needed by the staff to assembly this working group.

Question 11 [addressed to Douglas Broaddus, NRC]: Who are the members of the decommissioning transition working group?

Answer 11 [response from Douglas Broaddus, NRC]: Representatives from the NRC Offices of NRR, NSIR, NMSS, OGC and the Regions participate in the working group.

Question 12 [addressed to Douglas Broaddus, NRC]: Please elaborate on the Decommissioning Transition Working Group Final Report content and if/when it will be publically available.

Answer 12 [response from Douglas Broaddus, NRC]: The primary objective of the reactor decommissioning transition working group final report is to capture and document (knowledge management) the staff experience gained during a complete transition of a permanently shutdown power reactor from the NRR to NMSS. In addition, it will provide recommend long-term actions to improve the power reactor decommissioning transition process, such as development of guidance, rulemaking, and changes to policy or procedures. A determination as to whether or not the report will be publicly available has not yet been made.

Question 13 [addressed to Douglas Broaddus, NRC]: Please provide the timing of the Decommissioning Transition Working Group Final Report relative to the rulemaking.

Answer 13 [response from Douglas Broaddus, NRC]: The NRC staff working group expects to issue a final report sometime near the end of 2015 and should help inform the staff's efforts on a reactor decommissioning rulemaking which has a target completion date of 2019.

Question 14 [addressed to Douglas Broaddus, NRC]: What about other exemptions? There must be a whole host of Part 50 & 55 chapters no longer needed, e.g. 50.59, 50.65, Part 55. Is there something generic or does an individual exemption need to be submitted for each?

Answer 14 [response from Douglas Broaddus, NRC]: Most regulations in 10 CFR Part 50, as well as other Parts of 10 CFR are applicable only to a reactor that is authorized to operate. A reactor that has submitted certification of permanent cessation of operation and permanent removal of fuel from the reactor vessel in accordance with 10 CFR 50.82 is no longer authorized to operate. Therefore, many regulations in 10 CFR are no longer applicable based on the permanent shutdown and defueled status of the reactor. Otherwise, operating reactor regulations continue to be applicable to decommissioning reactors.

In the examples noted in the Question, 10 CFR 50.59 and 10 CFR 50.65 continue to be applicable and required throughout the reactor decommissioning until license termination. 10 CFR Part 55 applies to licenses for reactor operators. Upon NRC approval of a certified fuel handler training program, and amendment to the appropriate technical specification administrative controls in Section 5 of technical specifications, requirements in Part 55 are no longer relevant. Therefore, an exemption to requirements in Part 55 is not needed during reactor decommissioning to address changes made to operator staffing and training.

Question 15 [addressed to Douglas Broaddus, NRC]: It was stated that the transition to decommissioning has caused challenges because it is not automatic. Can you explain why this is a challenge?

Answer 15 [response from Douglas Broaddus, NRC]: Many regulations applicable to operating reactors continue to apply to reactors that are permanently shut down and are no longer authorized to operate. Because the risks at a permanently shut down reactor are significantly lower compared to the risks from operating reactors, application of certain operating reactor regulations to decommissioning reactors may not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule. In addition, compliance with certain regulations would result in undue hardship or other costs that are significantly in excess of those contemplated when the regulation was adopted, or that are significantly in excess of those incurred by others similarly situated. In these circumstances, the licensee may apply for an exemption to the regulation and the NRC staff must review and evaluate the licensee's request. This is often challenging to both the licensee and the NRC in that the resources, effort and time to obtain these exemptions are not as efficient, effective, or as transparent to the public as pre-existing regulations for decommissioning.

Question 16 [addressed to Douglas Broaddus, NRC]: Why does NRR have the lead for financial assurance when it is NMSS that will have to deal with the licensee if there are insufficient funds?

Answer 16 [answered Douglas Broaddus, NRC]: The NRC staff center of expertise on most financial matters involving reactor licensees resides in the NRC's financial analysis branch of the Office of Nuclear Reactor Regulation (NRR). The expertise of this branch includes, among many different facets of financial consideration of reactor licensing, the evaluation of the adequacy and sufficiency of a licensee's decommissioning trust fund. Regardless of which NRC organization has licensing oversight of a decommissioning reactor, the technical and regulatory evaluation of areas concerning financial assurance of the decommissioning trust fund are most efficiently addressed by the NRR financial analysis branch.

Question 17 [addressed to Marlayna Vaaler, NRC]: Are the pace and other aspects of decommissioning any different for nuclear power plants under Public Utility Company (PUC) oversight/control versus those that are not PUC regulated?

Answer 17 [response from Marlayna Vaaler, NRC]: Not directly. The pace and schedule of decommissioning at both regulated and merchant commercial power reactors is dictated primarily by the licensee / utility company depending on the decommissioning strategy they have chosen. The main difference between the two types of plants lies in the availability of additional funding for decommissioning activities. While both types of plants are required to possess decommissioning trust funds that will adequately cover the costs of radiological decommissioning at the site, public utility-controlled facilities may have access to additional fund sources from the public utility commission should they wish to speed up or otherwise change their overall decommissioning strategy or timeline.

Question 18 [addressed to Marlayna Vaaler, NRC]: The NRC allows any one of three "modes" for nuclear power plant decommissioning. Has the NRC conducted a general risk assessment on which mode is least risky in a probabilistic risk analysis (PRA) sense? Are plant-specific decommissioning PRAs conducted consistent with NRC "risk-informed" decision making?

Answer 18 [response from Marlayna Vaaler, NRC]: The NRC and licensees have performed PRAs for maintenance and refueling outages, of spent fuel pools and of spent fuel storage casks. We have not performed PRAs of permanently shut down and/or decommissioned plants.

Question 19 [addressed to Marlayna Vaaler, NRC]: Throughout the discussion with the public were there many adjustments that had to be made to provide the public assurance regarding the economic loss and environmental concerns of decommissioning?

Answer 19 [response from Marlayna Vaaler, NRC]: For discussions involving the socioeconomic and other environmental impacts of a nuclear power plant permanently shutting down and entering into decommissioning, the NRC staff uses the information contained in NUREG-0586, "Generic Environmental Impact Statement of Decommissioning of Nuclear

Facilities," Supplement 1, "Regarding the Decommissioning of Nuclear Power Reactors," to underpin conclusions in these areas. Specifically, this document states that the staff has considered available information, including comments received on the draft of Supplement 1 of NUREG-0586, on the potential impacts of decommissioning on socioeconomics. There are no NRC conclusions to be drawn from the socioeconomic information provided. NUREG-0586, Supplement 1 makes similar conclusions in numerous other environmental impact areas. The document can be viewed in its entirety on the NRC public website.

Question 20 [addressed to Tom Palmisano, SONGS]: Is funding for D&D "pledged" or approved and available for D&D of SONGS from all owners?

Answer 20 [response from Tom Palmisano, SONGS]: Under the California Nuclear Facilities Decommissioning Act of 1985 (CPU Code Sections 8321-30), the customers of Southern California Edison Company (SCE) and San Diego Gas & Electric Company (SDG&E) are obligated to pay for all decommissioning expenses that are deemed reasonable by the California Public Utilities Commission. These two investor-owned utilities own more than 95% of the decommissioning liability for SONGS 2 & 3 and 100% of the remaining decommissioning liability for SONGS 1. In addition, under the San Onofre Decommissioning Agreement that was executed by these two companies and by the Cities of Anaheim and Riverside in April 2015, all four decommissioning co-participants are contractually obligated to pay their respective shares of the SONGS 1, 2, & 3 decommissioning expenses.

Question 21 [addressed to Tom Palmisano, SONGS]: (a) What impacts on San Onofre plant during transitioning into decommissioning if there is a severe earthquake like Fukushima? (b) What is the expected cost for the decommissioning at the end of decommissioning and license termination?

Answer 21 [response from Tom Palmisano, SONGS]: (a) San Onofre spent fuel pools and dry cask storage systems are seismically designed systems, and would function whether the plant was operating, transitioning to decommissioning, or in dismantlement. San Onofre's spent fuel pools are steel-lined and structurally robust, with hardened steel-reinforced thick concrete enclosures. The spent fuel pools are seismically designed to withstand a peak ground acceleration of 0.67g. The racks in the spent fuel pools are designed to keep the fuel in its designated configuration during and after a seismic event. The water level in the pools is typically 23 feet above the top of the fuel rods. The spent fuel pool makeup water systems are also designed to withstand a peak ground acceleration of 0.67g. In the event the pool water level is reduced due to a seismic event, these redundant systems are designed to replenish any reduced water inventory. The dry cask storage system is also structurally robust and is designed to withstand a peak ground acceleration of 1.5g.

(b) The total estimated cost to decommission SONGS 2 & 3, as of the permanent retirement in June 2013, is $4.411 Billion (2014 dollars). This estimated cost includes all license termination, spent fuel management, ISFSI decommissioning, and site restoration expenses. The estimated cost to complete the decommissioning of SONGS 1 is $199.2 Million (2014 dollars). The scope of SONGS decommissioning includes the removal and disposal of all improvements from the site, and the decontamination of the site, as required by the site lease contracts.

Question 22 [addressed to Tom Palmisano, SONGS]: What is the driver for decommissioning in 20 years? It seems that you would want to be clear on what the end state requirements would be before starting.

Answer 22 [response from Tom Palmisano, SONGS]: While specific site restoration standards are still the subject of negotiations with the U.S. Department of Navy, the end state of returning the land to the U.S. Department of Navy for unrestricted use is clear and SCE believes the current schedule of decommissioning in 20 years brings the best balance for a safe, efficient and cost effective decommissioning.

Question 23 [addressed to Tom Palmisano, SONGS]: Considering the "brown outs" California was having while SONGS was operating, how is California going to make up the power with the loss of SONGS?

Answer 23 [response from Tom Palmisano, SONGS]: The closure of SONGS could create a demand gap of electrical power in South Orange County. In order to ensure electricity reliability remains at dependable levels, SCE launched a multi-year, comprehensive study designed to determine whether preferred resources -- including clean energy options such as energy efficiency, energy conservation, solar, wind, and energy storage -- can meet the constantly changing demands for electricity in the central Orange County area. Additional details can be found on https://www.sce.com/wps/portal/home/about-us/reliability/meeting-demand/our-preferred-resources-pilot/!ut/p/b0/04_Sj9CPykssy0xPLMnMz0vMAfGjzOK9PF0cDd1NjDzdgy1cDRy9TQOMLF0MDSwMDPULsh0VAZDpshk!/

Question 24 [addressed to Tom Palmisano, SONGS]: What are useful lessons from the successful decommissioning of Unit 1?

Answer 24 [response from Tom Palmisano, SONGS]: SCE learned many useful lessons from the successful decommissioning of Unit 1. SCE demonstrated that it could perform a large scale decommissioning project with a superior industrial safety record and while maintaining worker exposures to radiation below estimated levels. SCE demonstrated the successful use of many of the technologies that will be used in the decommissioning of Units 2 & 3. SCE attributes many of the successes from the decommissioning of Unit 1 to effectively benchmarking the other decommissioning projects that were in progress at that time, and from employing the use of industry experts in certain aspects of decommissioning. SCE has taken into consideration the lessons learned from the SONGS Unit 1 decommissioning in its planning for the Unit 2 & 3 decommissioning process. In addition, SCE learned that the material take-off calculations that were used in prior decommissioning cost estimates do not necessarily equate to the volumes of materials that must be shipped to radioactive waste disposal facilities, and therefore learned how to more accurately estimate low level radioactive waste packaging, shipping, and disposal costs.

Question 25 [addressed to Tom Palmisano, SONGS]: Tom Palmisano said there are "a lot" of challenges associated with the decommissioning of SONGS. He mentioned (1) laydown area and (2) politically charged environment. What are the other challenges now and ahead? Technical? Non-technical? Your lessons learned appear as common issues with decommissioning (reduce hazards, public engagement, and communication with regulators).

Answer 25 [response from Tom Palmisano, SONGS]: In the two years that San Onofre has shut down and started its decommissioning planning, the challenges are similar to those that other decommissioning plants face, including the lack of a permanent fuel repository. It is anticipated that many lessons will be learned and shared with the industry in the coming years.

Question 26 [addressed to Tom Palmisano, SONGS]: Stewardship - leave community better off. What exactly does that mean if you are planning unrestricted release?

Answer 26 [response from Tom Palmisano, SONGS]: The NRC allows for a decommissioned nuclear power plant site to be restored to “Restricted Use” or “Unrestricted Use.” Of the two choices, “Unrestricted Use” is more beneficial for the local community because regulatory controls are no longer required by the NRC (the residual radiation would be below NRC's limits of 25 millirem annual exposure). In other words, with “Unrestricted Use,” the land owner (such as the U.S. Department of Navy, in the case of San Onofre) would be able to use the land without any constraints, including farming. “Restricted Use” on the other hand, requires institutional controls, financial assurance and other restrictions on land use.

Question 27 [addressed to Tom Palmisano, SONGS]: The NRC has years of active engagement with the public regarding material and environmental concerns. Did you gain your insights from the NRC on the principles on engagement you use with the public?

Answer 27 [response from Tom Palmisano, SONGS]: Many insights for decommissioning, including the guiding principle of engagement with public and the concept of a Community Engagement Panel, were gained through benchmarking other decommissioned and decommissioning nuclear plants, as well as the NRC’s experience.

TH30 Recent Operating Reactors Materials and Mechanical Component Issues

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The questions below were not answered during the above session.

Question 1 [addressed to DeLisa Pournaras, SNC]: Would you discuss whether the Hatch operating experience of core shroud cracking suggests a potential need to revise neutron fluence thresholds for susceptibility to irradiation-assisted stress corrosion cracking and irradiation embrittlement?

Answer 1 [response from DeLisa Pournaras, SNC]: The Hatch Unit 1 shroud results do not suggest a need to change any fluence-related thresholds. In fact there is not a "defined threshold" for the onset of IASCC in a core shroud nor do inspection criteria hinge on any such threshold. Fluence thresholds are used to determine crack growth rates to be used as well as fracture toughness guidance and flaw evaluation methodology (limit-load, LEFM or EPFM). Existing BWRVIP documents provided all the guidance Hatch needed regarding fluence to assess the indications. As a member of the BWRVIP, we are aware that the industry is continuing to collaborate on irradiated materials testing through the BWRVIP, including the Hatch shroud boat sample testing. This is a normal part of the industry's ongoing effort to fully understand and respond appropriately to any industry operating experience and to prepare in advance for long term operation. Hatch remains committed to support those efforts.

Question 2 [addressed to Glenn Gardner, Dominion Generation]: There was a recent example of baffle jetting in a plant where this was not expected. One possible cause is void swelling.

a. How will the inspection guidelines be modified to account for this experience?
b. Are there methods that can be used to detect distortion or jetting before fuel damage occurs?

Answer 2 [response from Glenn Gardner, Dominion Generation]:

a. There are no immediate plans to revise the guidelines for baffle jetting, however all OE is noted and we expect to discuss baffle jetting during industry meetings this year. The baffle jetting is not a significant safety issue and is detectable almost immediately if minor fuel damage occurs; the typical corrective action is a conversion to an up flow design, which eliminates the jetting.
b. The MRP-227 inspections of baffle plates is for visually observable variations in baffle seam gaps, primary those that might be due to void swelling, and are not intended to detect the potential for baffle jetting. Development of a reliable predictor for baffle jetting (apart from the known susceptibility factor of being a down flow design) does not appear to be feasible. Gaps in the mechanical joint seams between baffle plates are small to begin with. The inside corner joint gaps in particular, as was the case in the recent OE, are not amenable to meaningful visual inspection.

Question 3 [addressed to Glenn Gardner, Dominion Generation]: When will additional investigations to evaluate the weld & CASS indications be conducted?

Answer 3 [response from Glenn Gardner, Dominion Generation]: The evaluation of the assumed worst case regarding the indications is complete and has acceptable results. Further investigation, by additional inspection, would entail confirmation of indication status or potential reassessment of the indication(s) as being non-relevant, non-aging related features such as scratches. The schedule of this activity is at the plant's discretion but I believe it will occur before the next required MRP-227 inspection.

TH31 Seeking a Path Forward – Digital in Nuclear Plant Safety Systems

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Session Coordinator: Karl Sturzebecher, Electronics Engineer, Division of Engineering, NRR/NRC, 301-415-8534, Karl.Sturzebecher@nrc.gov

The questions below were not answered during the above session.

Question 1 [addressed to NRC]: Elaborate on use of modern digital I&C for plant protection / safety systems of Vogtle & N.C. Summer: 1) Elaborate on anything you can say about I&C for SMRs. 2) What do SMR must measure for control & safety of what type of (?lessons) is available for SMR, do new (?lessons) have to be developed for SMRs?"

Answer 1 [response from NRC]: Vogtle Units 3&4 and V.C. Summer Units 2&3, when completed, will be the first, nearly all-digital plants in the U.S. The AP1000 is a passive plant design. AP1000 digital instrumentation and controls (DI&C) is designed to support a passive plant design and some aspects of the I&C architecture are distinct from other design centers. The DI&C design features a number of design attributes that will enhance or improve system reliability, ease of maintenance, fault tolerance as well as featuring modern human machine interfaces in the main control room design. Westinghouse Common Q digital platform is used for the safety I&C, and the main control room is essentially all digital, highly integrated, and based on rigorous human factors engineering.

Regarding small modular reactors (SMRs), the staff is developing design-specific review standards (i.g., guidance specific to individual SMR designs) as a means to enhance the safety focus and streamline the staff's licensing reviews taking into account the uniqueness of SMR I&C design features and characteristics. The staff has been interacting with the potential applicants regarding their I&C designs that are being developed. No SMR design certification application has been received for staff's review.

Question 2 [addressed to NRC]: 1) What are the TOP three technical issues? 2) What is the NRC acceptance requirement for the technical issue?

Answer 2 [response from NRC]: To answer the 2nd question first, recognizing that new technologies can present challenging levels of complexity, the NRC staff seeks to consistently use the regulatory requirements found in 10CFR50, which includes industry consensus standards incorporated by reference, such as IEEE603, as well as the review criteria found in the NUREG 0800 Standard Review Plan (SRP) and guidance found in associated Regulatory Guides. The Office of Nuclear Regulatory Research supports the staff in meeting these challenges by updating regulatory guidance, Branch Technical Positions and Interim Staff Guidance.

One of the main issues with the changing technology has been describing the requirements with a clear interpretation of the relationship to regulatory acceptance requirements. There are a number of technical issues that continue to be challenging, among these issues are:

1. What must be considered by staff to support a conclusion that a proposed safety system design adequately addresses the potential for software common cause failure?
2. How best to address complicated dependencies associated with digital communications, and
3. How to identify better methods for evaluating the hazards posed by faulty software requirements specifications in digital systems.

For staff to achieve reasonable assurance, it must understand: the functional requirements for the given class of digital technology; how the systems function, including their communications interconnectivity; and the way(s) the given technology can fail. The quality of any proposed system depends on how effectively and rigorously the licensee has implemented the software life cycle process, which includes the conduct of a “D3” (Diversity and Defense in Depth) analysis. Such an analysis needs to provide an assessment of ways to mitigate and cope with digital failures at any point/level in the architecture of the digital system.

For new reactors, the presentation addressed the most difficult digital instrumentation and controls (DI&C) technical issues that applicants faced during recent design certification reviews. Among these issues are:

1. Independence – Applicants had varying levels of difficulty in adequately demonstrating that independence requirements have been addressed (based on IEEE Std. 603-1991, Clause 5.6 which is incorporated by reference into 10 CFR 50.55a(h)). Specifically, applicants had difficulty in demonstrating sufficient independence when incorporating bi-direction data communication between safety-related divisions and between safety- and non-safety-related I&C systems.
2. Postulated Failures in Non-safety-related DI&C – Much of the non-safety-related DI&C for the various design certifications are highly integrated. Specifically, multiple control functions are often integrated under a single common platform or distributed control system. This design approach brings along characteristics such as commonalities, dependencies and systems interfaces that need to be considered when assessing the potential for failure(s) in the non-safety-related DI&C to affect the safety of the proposed

plant design (i.e., exceed the plant's safety analysis) or affect the ability of the safety-related I&C systems to perform their required functions.

3. Incomplete or Inconsistent Design Information – During new reactor licensing reviews; the staff found examples of supporting design information that was incomplete, unfinished or missing. There were also a number of occurrences where the staff found inconsistencies between the DI&C design and the interfacing plant systems, such as errors in cross-referencing different chapters of applicants' final safety analysis reports (FSARs). Incomplete or inconsistent design information within design documentation has resulted in delays in reviews because the information in question must be provided and/or clarified by the applicants before the design aspects in question can be evaluated. Incomplete or inconsistent design information adds uncertainty to the review process as some design claims cannot be verified if supporting design detail has not been provided.

Question 3 [addressed to all]: How do you address all the "small" digital systems, e.g. firmware? We have recognized in e.g. relay. Sometimes you even do not know that "code" is included on ASIC's, PLD's, FPGAS.

Answer 3 [response from Ray Herb, Southern Nuclear Operating Company]: Generally, we assess the performance of the component and the potential hazards associated with that component failure. This largely depends on the qualification of the component, is it safety related or non-safety related, the effort to verify the component firmware will depend largely on that and the potential hazards associated with the component. In short, if the firmware performs a safety function, it will have to be inspected and reviewed quite extensively, and in some cases rejected, especially if it cannot be verified through sufficient testing and code inspection.

Answer 3 [response from NRC]: The NRC is aware of the increasing use of digital technology in plant equipment, which may contain software, software developed firmware, or software developed programmable logic.

The NRC is developing a Regulatory Issue Summary (RIS) that addresses the use of embedded digital devices in safety-related systems. A draft of the RIS can be found using the Agencywide Documents Access and Management System (ADAMS) Accession No. ML13338A769. The final RIS is expected to be issued by mid-2015.

Question 4 [addressed to NRC]: Digital Licensing – You discussed that interfaces between safety and non-safety systems are adding to complexity of reviews. Do you see cybersecurity rules adding too reducing complexity of licensing?

Answer 4 [addressed to NRC]: NRC is aware of the interfaces of safety and non-safety systems, and cyber regulations. For safety systems the applicant needs to demonstrate the independence from non-safety systems, per IEEE Std. 603-1991, Clause 5.6, which is incorporated by reference into 10 CFR 50.55a(h)). The cyber security requirements are addressed under 10 CFR Part 73.54, "Protection of digital computer and communication systems and networks." The effect of the cyber security criteria on the system complexity depends on the number and types of digital device interfaces included in the design. With no interfaces then the cyber security involvement is limited. If a digital device is identified as Critical Digital Asset (CDA), then the applicant addresses the CDA in accordance with Regulatory Guide 5.71; "Cyber Security Programs for Nuclear Facilities."

Question 5 [addressed to NRC]: The discussion included challenges with hazards in non-safety digital I&C design. The differences in the level of detail in the design description and ITAAC between safety and non-safety systems is significant.

1. Can NRO provide insights on what drove the differences and why it was acceptable?
2. What practices used during new plant licensing of digital I&C could be used to streamline regulatory reviews & reduce regulatory uncertainty?

Answer 5 [response from NRC]: The differences in level of detail between safety DI&C and non-safety DI&C design is mostly rooted in the differences in regulatory requirements placed upon systems or functions of differing safety class. In other words, the level of detail for design descriptions in the FSAR or ITAACs is generally commensurate with the safety class of the systems, structures or components in question. The requirements applicable to safety systems (such as 10 CFR Part 50 Appendix A or Appendix B) necessitate a higher level of detail to adequately demonstrate compliance in terms of quality of design and fabrication of hardware/software, system performance, etc. For non-safety DI&C, the safety concerns principally revolve around the potential effects of undesired behavior of non-safety I&C systems on plant safety or the performance of safety functions. To this end, the level of detail would be more specific. For example, the Appendix B Quality requirements are not applied to non-safety DI&C because these systems are not relied upon to perform a safety function.

As stated during the presentation at the RIC, for new reactor licensing the staff is always looking for new ways to streamline the review process. One of the lessons learned from recent reviews is to place a greater emphasis on the pre-application and acceptance review process. The goal being to identify, as early as possible, any problem areas, missing design information, etc. that can adversely affect the formal review process. The earlier these types of issues can be identified, the earlier the staff and applicants can work to resolve these issues before the application is docketed. This can reduce the number of requests for additional information and help ensure that there is a common understanding and expectations between staff and applicants. A streamlined review process will lead to fewer scheduling delays and ensure a more timely completion for reviews.

Question 6 [addressed to NRC]: Wendell Morton "Can you address the timing of NRC I&C reviews for new plants, with particular reference to ITAAC for I&C systems? Thank you."

Answer 6 [response from NRC]: For new reactor licensing, ITAACs are a required portion of the design certification as per 10 CFR 52.47. ITAACs for new reactor I&C are reviewed concurrent with all other aspects of the I&C design during the staff evaluation of the design certification application. The closure of the ITAACs occurs as the licensee's post-licensing activities prior to the fuel load. Through inspection, the NRC staff carefully selects and verifies the adequate closure of ITAACs before the fuel load is allowed.

Question 7 [addressed to NRC]: Why not have digital venders certify and license equipment for specific applications?

Answer 7 [response from NRC]: This is a vendor decision. Some vendors have submitted applications for licensing of application specific equipment. The NRC has evaluated several such topical reports. Examples include: GEH NUMAC Power Range Neutron Monitoring System, and the Caldon Ultrasonic Flow Measurement system.

The Westinghouse Common Q Topical Report also includes several application specific annexes. More recent submittals of digital platforms have been generic with no reference to any specific application.

Question 8 [addressed to NRC]: What specific actions is the NRC taking to eliminate the regulatory uncertainty for licensing digital safety systems?

Answer 8 [response from NRC]: The term Regulatory Uncertainty is a perceived notion of unknown difficulties a licensee may face during a licensing process. Though it is not possible to eliminate uncertainties outside of the context of a specific design, the NRC has taken several steps to reduce these uncertainties associated with the licensing of digital safety systems.

The review guidance used by the NRC technical staff (NUREG 0800 Chapter 7) is publicly available and the NRC is currently revising it to correct errors, add clarifications and to address changes made to its referenced standards since it was last updated in 2007.

The NRC worked with the nuclear industry to develop Interim Staff Guide Digital I&C-ISG-06, "Licensing Process" to clarify the Standard Review Plan SRP criteria and to establish a defined process for licensing digital I&C safety systems in accordance with existing regulations. ISG-06 describes the licensing process that may be used in the review (against licensing criteria – the Standard Review Plan, NUREG-0800) of license amendment requests associated with digital I&C (I&C) system modifications in operating plants originally licensed under Part 50.

ISG-06 is currently being piloted as part of the Diablo Canyon PPS system replacement license amendment.

Question 9 [addressed to NRC]: Regarding 50.59 uncertainty is there an ISG 6 Phase '0' type vehicle industry can use during 50.59 process?

Answer 9 [response from NRC]: The NRC welcomes and encourages licensees to schedule pre-submittal or phase 0 meetings to discuss 50.59 criteria as it pertains to a proposed digital safety system design. Though the phase 0 meeting concept as defined in ISG-06 is intended to apply to subsequent license amendments, it can also be used for prospective 50.59 plant modifications or to aid in the assessment of 50.59 criteria.

Question 10 [addressed to NRC]: NRC does not seem to acknowledge that licensees will not pursue digital upgrades until the regulatory process is stream lined and more predictable. They cannot wait for more licensees to come and try out the process. Delayed regulatory approval can impact plant outages and costs. The regulatory process is not beneficial to safety. Does/will NRC plan to revise ISG-06 in the near term?

Answer 10 [response from NRC]: Revising ISG-06 is an option, and may be one that staff exercises to achieve a more timely, short term update; however, it would be preferable to incorporate the lessons learned from ISG-06 into a more permanent form of guidance. Current options being considered are:

- Add a new sub-chapter to NUREG 0800 Chapter 7 to include new guidance from ISG-06.
- Revise existing SRP Chapter 7 sections to incorporate the concepts of ISG-06.
- Create a new Branch Technical Position which would be a revised version of the existing ISG-06.

A public workshop will be held later in 2015 to discuss lessons learned from the Diablo Canyon ISG-06 Pilot project and to evaluate the available paths forward.

Question 11 [addressed to NRC]: Regulatory clarity is being challenged due to blurring of lines between the following areas: (1) 50.59 screen & evaluation process and associated technical documentation, (2) Quality requirements – implying the use of safety related rules for non-safety systems, and (3) Cyber security. What efforts are occurring within the NRC to focus these efforts across NRC branches and produce succinct & coordinated positions in these areas while maintaining appropriate separation?

Answer 11 [response from NRC]: There are internal instructions to ensure technical consistency across associated technical review branches.

Industry has and should continue to use NEI to coordinate the communication of any perceived "blurring of lines" to allow a success path to be crafted by the NRC and Industry representatives. For example, NEI and some of its utility members have engaged in a detailed effort to evaluate and improve the criteria in NEI 01-01 for the conduct of 10CFR50.59 screenings and evaluations. NRC has been engaged with NEI and EPRI and the utility representatives for the last year and a half on this topic. Staff has been informed by NEI that results of this effort will be forthcoming later this year.

Question 12 [addressed to all]: Is ISG-6 working? Has it helped with submitted information? What can we do to improve industry level or financial commitment vs. degree of regulatory uncertainty?

Answer 12 [response from NRC]: Yes, the ISG-06 process has clarified many aspects of the licensing process for digital I&C systems. Some areas of the guidance have proven to be more beneficial than others. Here is a list of areas of ISG-06 which have been beneficial during the Diablo Canyon PPS project:

- Two phase document submittal process.
- Annex B – List of information to be provided to support a license amendment.
- Use of Phase 0 meetings to identify potential design and licensing issues prior to license amendment submittal.
- Use of Sharepoint to provide NRC staff with opportunity to preview documents prior to submittal and to audit support documents that do not need to be docketed.

A public workshop will be held later in 2015 to discuss lessons learned from the Diablo Canyon ISG-06 Pilot project and to evaluate the available paths forward.

Answer 12 [response from Ray Herb, Southern Nuclear Operating Company]: We have not used the ISG-6 process. We have adopted at Southern Nuclear a wait and see position with respect to ISG-06, instead we have concentrated on changes that do not trip the requirements for a licensing change with respect to digital.

Answer 12 [response from Jay Amin, Luminant- Comanche Peak Nuclear Power Plant]: No, ISG 6 will not work and licensees will shy away from performing any SR upgrades requiring an LAR submittal unless the process is changed. Several of my slides amplifies the issues including recommendations

No it has not helped the majority of the industry since too much information is required; the process and costs negate any benefits; We need up-front regulatory certainty prior to us spending major portion of budgets only to find out we have to double the budgets and extend the schedules; I do not believe most licensees would take this path, it would be cheaper to look at alternatives. We have to exercise prudency and projects have to be cost justified.

The amount of details required with submittal of the LAR, coupled with exhaustive staff reviews and unknowns add regulatory risk and uncertainty For example, the LAR approval comes *after* we have expended significant resources.

The LAR process (ISG 6) simplification should focus on system design and demonstration of compliance to regulatory criteria. Safety determination should be made at the system design phase similar to new plants, with an SER issued with open items requiring specific NRC audits and inspections for confirming compliance in lieu of detailed design reviews.

LAR would submit SPECIFIC System Architecture, D3 analysis, including coping analysis and all the digital I&C design details (specs, design, V&V, etc.) would be available for audit and/or inspection during System/Design Modification Life Cycle Development.

The above approach will now provide clear success path and removes regulatory uncertainty. If the licensee does not address what he/she committed, then the NRC inspections/audits will identify this during the life cycle (LC) phase and can be corrected at that time rather than after the design is complete. Thus the burden of proof lies with the licensee and removes the regulatory uncertainty at the back-end since the success path is drawn and agreed upon up-front in the process.

Question 13 [addressed to Jay Amin, Luminant- Comanche Peak Nuclear Power Plant and NRC]: Regarding Jay Amin slide which said – “Failure to adopt current technology…In what way can the NRC do a better job with respect to reducing regulatory burden on the near term?

Answer 13 [response from Jay Amin, Luminant- Comanche Peak Nuclear Power Plant]:

Near Term:

1. Need to quickly clearly define when an LAR is required; this is very important to both the industry and the NRC. This requires addressing allowing licensees to credit prevention/mitigation strategies in addressing CCF and reaching a conclusion of low likelihood of CCF.
2. Initiate pilots with the industry that would test the issuance of the SER upfront based on submittal of Licensee Specific Architecture and D3 analysis
3. Focus clearly on the Criteria and ensure it is clearly defined for digital
4. NSIR regulated security including cyber security. NRR/NRO has to drive the criteria as to what is a CDA and what is not a CDA based on SSEP functions that if compromised will result in adverse impact to safe shutdown or results in radiological releases, or result in greater than 300MWe power reduction. IT is NRR/NRO function to define the criteria and then NSIR can ensure that this criteria is implemented and security controls addressed

We must have a practical approach to CGD for I&C equipment. The industry is facing obsolescence with many I&C components. The replacements are commercial off the shelf I&C components that are used in thousands of applications. However, for us to use them in a SR application, they have to go thru the same process as if we were custom designing a new product, qualifying the product, (meet ALL regulatory criteria) rather than crediting operating experience and other value added digital attribute only such as SQA Process. At the end of the day, we spend thousands of dollars on paper that did not result in any single change to the product. So the current process is not cost effective.

Answer 13 [response from NRC]: The NRC is open to suggestions from industry or the public on ways to improve the regulatory review process as long as appropriate levels of safety can be maintained. All such suggestions will be discussed in an open format during the public workshop to be scheduled in fall of 2015.

Question 14 [addressed to Jay Amin, Luminant- Comanche Peak Nuclear Power Plant]: Are licensees ready and mature enough to deploy digital I&C for safety systems with you proposed approach, which is LAR early in the life cycle for the upgrade project?

Answer 14 [response from Jay Amin, Luminant- Comanche Peak Nuclear Power Plant]: I understand the context of this NRC question based on known industry issues. However, this is not the whole industry. Transition from analog to digital is already occurring successfully in the non-safety side.

There is a good understanding of digital and licensees are upgrading their processes to get even better. As licensees are transitioning from analog to digital, they are applying lessons learned and sharing experience to get even better so that mistakes are not repeated.

I strongly believe the licensees are fully prepared for this, give them a chance. Also the good thing about this process is that the NRC gets to pre define where they would like to inspect/audit in the process and this can be built into the licensee project schedules, so the safety risk is minimized.

Question 15 [addressed to Jay Amin, Luminant- Comanche Peak Nuclear Power Plant]: What do you for NRC to do for digital I&C implementation? Are you talking about digital I&C for safety or non-safety; was your focus on plant protection systems I&C or all I&C?

Answer 15 [response from Jay Amin, Luminant- Comanche Peak Nuclear Power Plant]: Yes, I am talking about Digital I&C for Safety Systems. The focus needs to be SR Digital upgrades and CGD of digital as a priority. Secondly, if the licensee is performing NS upgrades using a single network based approach, then per their FSAR commitments (LBDs and Ch. 15), they will have to reanalyze the approved analysis and perform 50.59 reviews to determine if an LAR is required or not. For other upgrade techniques, the current process in place is good if the issues identified in my presentation are addressed.

Question 16 [addressed to Jay Amin, Luminant- Comanche Peak Nuclear Power Plant]: If NRC changes its process and approves topical reports earlier in the process (as noted on your slide) would this create regulatory uncertainty as part of the NRC inspection process? Would this obviate the need for a 50.59 process for topical reports?

Answer 16 [response from Jay Amin, Luminant- Comanche Peak Nuclear Power Plant]:

Good questions:

1. If a pre-qualified platform was used, the licensee submits the specific architecture and the D3/coping analysis; the NRC can work with the licensee to ensure clear cut agreements are reached. This would ensure licensee understanding what is expected of him/her up-front before they spend majority of their budgets. I believe an up-front agreement is required; the licensee may choose to go forward with the upgrade or may decide that it is not path that they want to take.

2. Once NRC approves the submittal and issues an SER, both parties know what is expected of each other. Since this is done up-front, now there is no misunderstanding and if there are issues, they are caught during the modification LC phases which are much easier to deal with than they are at the back end (Current LAR Process).
3. No it does not prevent the need for a 50.59 process for a topical report. The vendors can still independently submit their platforms for topical report which is then used by the licensees as a starting point. However, in my presentation I touched the point that vendors need to be afforded a process similar to 50.59 so they can keep the platforms current rather than the current process which also adds to regulatory uncertainty and costs to licensees.

Question 17 [addressed to Jay Amin, Luminant- Comanche Peak Nuclear Power Plant]: Re: ISG-06 process: Would you prefer a 10CFR 52 like process?

Answer 17 [response from Jay Amin, Luminant- Comanche Peak Nuclear Power Plant]:

Yes, but the process must be well defined and jointly AGREED UPON BY BOTH PARTIERS SO THERE IS NO REGULATORY UNCERTAINTY. Since humans are involved, we need to ensure that we focus on acceptance criteria in practical terms. We must focus on items that are really value based, since if everything is treated as important, then the costs also make the upgrade non justifiable. Whether we like it or not, cost does matter now more than it did before since the power generation industry has changed in US.

Question 18 [addressed to Jay Amin, Luminant- Comanche Peak Nuclear Power Plant and Ray Herb, Southern Nuclear Operating Company]: Do older NPPs intend to adopt applicable codes + standards into their design + licensing basis when they adopt digital systems under 10CFR50.59? If so, how do they intend on accomplishing this?

Answer 18 [response from Jay Amin, Luminant- Comanche Peak Nuclear Power Plant]: Typically, they will follow the committed licensing basis in their FSAR for modifications to the existing digital systems.

If they are replacing the entire analog or digital systems, they will comply with the current standards for the specific upgrades in their design basis, and may even update their licensing basis if they choose to do so for newer regulations or later revisions of already committed standards.

This would have to be done since for example: If segregated I&C Analog systems such as NSSS control system was upgrades from analog to digital, one must address the effects of combining segregated I&C into one controller or few controllers, including credited NS I&C in Ch. 15. Few licensees have already done this successfully in the NS arena.

Answer 18 [response from Ray Herb, Southern Nuclear Operating Company]: SNC when making changes will follow the required codes and standards for all changes, regardless of digital, or LAR vs. 50.59. New systems must follow the applicable standards (e.g. IEEE 603, 7-

4.3.2) regardless of previous committed standards for existing analog systems, this is part of our design process.

Question 19 [addressed to Ray Herb, Southern Nuclear Operating Company]: What is or please define the stated 100% testing requirement of BTP-7-19?

Answer 19 [response from Ray Herb, Southern Nuclear Operating Company]: BTP 7-19 defines 100% testing as “every possible combination of inputs and every possible sequence of device states are tested and all outputs are verified for every case (100% tested).”

Question 20 [addressed to Ray Herb, Southern Nuclear Operating Company]: There is not much discussion of HFE for digital upgrades. Do you believe this is important?

Answer 20 [response from Ray Herb, Southern Nuclear Operating Company]: HFE is very important, and it should be addressed early in the development, many of the benefits of a digital control system are stranded without taking advantage of the potential human factors improvements like alarm presentation systems, computerized procedures and predictive maintenance. Usually there is a “seminal event” that drives human factors changes in every industry, like Three Mile Island did back in the early 80s; I think the movement to digital control systems will be that impetus for change for the next leap forward in control of nuclear power plants going forward. HFE is very important.

Question 21 [addressed to Ray Herb, Southern Nuclear Operating Company]: Often industry opinion regarding the NRC’s I&C licensing process are generated from applicant specific issues that then become perceived as generic road blocks by the industry. How can the NRC better communicate on licensing process concern with the industry?

Answer 21 [response from Ray Herb, Southern Nuclear Operating Company]: The nuclear industry is a learning organization; we have set up INPO specifically for this purpose. Generally we learn from the experience each licensee has with the NRC, it is our culture. Probably the biggest impact the NRC could have on that front is consistency, consistency within the regions, and between the region and between the regions and NRR and NRO.

Question 22 [addressed to NRC]: Reasonable assurance – Is 100% testing to include all possible system states considered a reasonable assurance threshold when this degree of testing generally cannot be performed?

Answer 22 [response from NRC]: “100% testing” is a reference to Branch Technical Position (BTP) 7-19, Section 1.9, “Design Attributes to Eliminate Consideration of CCF” in which the position states the following under Criterion 2, “Testability”,

> “A system is sufficiently simple such that every possible combination of inputs and every possible sequence of device states are tested and all outputs are verified for every case (100% tested).”

The portion of the Testability criterion to emphasize is that the system under question is "sufficiently simple" such that the described testing can be performed. If the system under question cannot be feasibly tested to this degree, then it would not be considered a simple system in terms of functionality, design, etc. If the system under question then is not sufficiently simple, then the argument cannot be made that consideration of software common cause failure (SWCCF) can be eliminated, using this Testability threshold. Performing 100% testing of a system is one acceptable method described in guidance. The other method is the built-in diversity through design. BTP 7-19 is staff guidance (versus requirements), and alternatives to the criteria in the guidance could be found acceptable as long as they are adequately justified. BTP 7-19 (Revision 6), Section B.1.9, speaks to the application of many system design and testing attributes, procedures, and practices that can significantly reduce the probability of common cause failure (CCF).

Question 23 [addressed to NRC]: Oconee implemented digital RPS/ES at all three of its units. The staff requested significant amounts of information during the review, including the software code. How does the NRC balance the need for reasonable assurance and the desire for absolute assurance in conducting its review?

Answer 23 [response from NRC]: The NRC did not perform a design review of the software used in the Oconee license amendment. A limited scope code review was performed in specific areas where the staff felt it was necessary to attain a greater level of design level understanding to establish a basis for its reasonable assurance determination. The software code for the Oconee RPS/ESPS application was not submitted to the NRC and the code reviews were performed within the context of audits conducted during the evaluation.

The NRC recognizes that resources needed to achieve absolute assurance in conducting its safety evaluations would be prohibitive. This is why the NRC instead uses the concept of "reasonable assurance". The NRC is committed to make improvements its review guidance as we gain experience with the use of digital technologies for I&C safety systems. The development of ISG-06 is an example of this type of licensing review process improvement, which we should note was developed in concert with industry. The NRC plans to use the improved evaluation processes for all future licensing reviews of digital I&C safety systems.

Question 24 [addressed to NRC]: The airlines have demonstrated protecting the health & Safety of public w/ installation and operation of digital controls into aircraft that performs "vital" high risk critical flight procedures (see ex) on a tremendous scale. Why has the NRC not learned from and advanced digital applications, on the same scale? What has the NRC done to learn from advanced aviation applications to advance nuclear upgrades (ex: landing aircraft under extreme weather conditions (e.g. solid fog))?

Answer 24 [response from NRC]: The NRC has learned from the submitted digital applications. For example: ISG-06 was a result of learning from the experience of reviewing license amendment applications. A design-specific review standard (DSRS) for the mPower project was a result of learning from the experience of applications for design certification. In response to the Advisory Committee on Reactor Safeguards, the Office of Nuclear Regulator Research pursued an expert clinic to learn from advanced digital applications. To date three research information letters demonstrate products of resultant learning from digital applications in other safety domains: RIL-1001 "Software-Related Uncertainties in the Assurance of Digital Safety Systems—Expert Clinic Findings, Part 1" [Agency Document Access and Management System (ADAMS) Accession Number ML111240017]; RIL-1002, "Identification and Analysis of Failure Modes in Digital Instrumentation and Controls (DI&C) Safety Systems-Expert Clinic Findings, Part 2" (ADAMS No. ML14197A201); and RIL-1101, "Technical Basis to Review Hazard Analysis of Digital Safety Systems" (ADAMS No. ML14237A359).

The decision to implement nuclear upgrades is up to the licensees. The NRC is responsible to review nuclear applications for safety. Good regulatory practices dictate that the NRC not select or promote particular techniques or technologies. Should an application propose a modification with technical basis in advanced aviation techniques, then the USNRC would review that application against applicable acceptance criteria. In contrast, the Department of Energy could establish activities to promote nuclear advancements including application of advanced aviation techniques.

The NRC had arranged presentations and seminars by flight control experts to which members of the IEEE/NPEC/SC6 standardization subcommittee and EPRI staffs were invited. NRC's research information letters RIL-1001 and RIL-1101 include information from the aviation industry (as well as other leading application sectors). The next research plan under development includes learning from the latest standards and direction of the aviation industry, as well as other sectors using digital technology. Some notable examples of experts engaged include:

- Dr. Darren Cofer of Rockwell Collins, leading avionics supplier
- Bruce Lewis, aviation systems researcher for the US Army Aviation and Missile Command Research Development and Engineering Laboratory (AMRDEC), Software Engineering Directorate
- Professor John Knight, performing civil aviation research for NASA
- Dr. Peter Feiler, SEI's (Software Engineering Institute) developer of the language, AADL

Question 25 [addressed to all]: How does the success and experience of other industries with "digital" being leveraged into nuclear power utilization, strategies, and development. Is there a big jump from non-nuclear to nuclear…mainly "Regulatory"?

Answer 25 [response from NRC]: The scale to which the NRC can put its learning to use is ultimately limited by the number of nuclear industry applications for digital safety systems. There has, as well, been a lack of "digital" applications in other industries that address regulatory requirements similar to those found in nuclear. This lack of comparability also limits the applicability of these experiences and the learning we could gain."

Answer 25 [response from Ray Herb, Southern Nuclear Operating Company]: We often use EPRI (Electric Power Research Institute) as our partner in evaluating experience from other critical industries that are subject to similar regulation and high levels of assurance, but we must be careful that the "success and experience" in digital technology is applicable to the unique requirements of nuclear power. The Regulator will allow us to use and will welcome this fresh look, but they do insist that the experience from other industries be justified for relevance to the nuclear industry. A good example is the NIST 800 controls which are the basis for the current NRC Cyber Security requirements; these were not created in the nuclear industry.

Question 26 [addressed to Ray Herb, Southern Nuclear Operating Company and Jay Amin, Luminant- Comanche Peak Nuclear Power Plant]: Are DI&C upgrades safety improvements, and if so, how as this been quantified? Does it consider any uncertainties?

Answer 26 [response from Ray Herb, Southern Nuclear Operating Company]: This is a tough question, because it cannot be answered directly with a quantified number. All changes to the plant controls digital or otherwise must be evaluated based on the merits of the change. All changes must be equal to or better than the original. Digital has the potential, if properly designed and implemented to be more fault tolerant and reliable, which can be a significant safety improvement, however, it can also be the converse. That is why each change must be implemented in a controlled and careful process that evaluates all the potential hazards and adjusts the design as needed to achieve the desire goals of simple and reliable. Since the goal is "as good or better" this does not lend itself to further quantification, in addition since there is no consensus method for establishing that quantification that has been approved for use in nuclear we typically compare digital upgrades to the currently installed systems. Time will tell, as we gain experience with digital, it may lead to a time when that quantification can be established through documented operating experience.

TH32 Future Direction of International Research for Reactors and Fuel Cycle Safety (Part 1)

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The questions below were not answered during the above session.

Question 1 [addressed to Won-Pil Baek, KAERI]: What is the difference between APR+ and APR 1400?

Answer 1 [response from Won-Pil Baek, KAERI]: APR+ maintains the basic design characteristics of APR1400 but incorporated several design improvements for enhancement of safety and economics. Design changes or improvements include:

- Capacity increase from 1,400 to 1,500 MWe with the increased number of fuel assemblies
- Passive Auxiliary Feedwater System (PAFS) instead of the auxiliary feedwater system of APR1400
- Advanced safety injection system features for more effective utilization of the emergency core cooling water
- Enhanced independence of safety injection trains
- Overall optimization of systems, components and structures, etc.
- More importantly, design, analysis and verification processes were implemented with maximum utilization the independent technology of the Korean industry.

Question 2 [addressed to Won-Pil Baek, KAERI]: Is the large-scale FCI facility in operation?

Answer 2 [response from Won-Pil Baek, KAERI]: Yes, we are still operating and will maintain the operability of the TROI facility. Current focus is given to the FCI behavior in case of the external reactor vessel cooling for in-vessel retention of molten corium.

Question 3 [addressed to Won-Pil Baek, KAERI]: Can you share with us your activity on the Severe Accident Management Expert System (SAMEX) development?

Answer 3 [response from Won-Pil Baek, KAERI]: Dr. Baek did not have any information to share about the Severe Accident Management Expert System (SAMEX) development.

Question 4 [addressed to Qinghua Zhang, NNSA]: Could you say a bit more about your research on evacuation planning? Do you develop evacuation time estimates as in the U.S.? What is the current baseline for EPZs in China?

Answer 4 [response from Qinghua Zhang, NNSA]:

1. The planning is called the virtual reality simulation of the population evacuation around the NPP. The target of the planning is to calculate the evacuation times under a variety of conditions. The planning is to develop a computer system used to imitate the evacuation process and calculation of ETE of population around the NPP. The system will include four functions, i.e., evacuation planning, evacuation process imitation, evacuation process demonstration and statics of key data.
2. We referenced the method used by US in developing and analyzing the evacuation time estimates in our planning, such as the constitutions of the evacuation time and the analyzing method of key factors.
3. On-site nuclear accident emergency planning for nuclear power plant should include plume emergency planning zone and ingestion emergency planning zone. The inner

zone of about 3-5km would be
established for the plume exposure pathways and an outer zone of about 7-10km would be established for the plume exposure pathways. A zone of about 30-50km would be established for the ingestion exposure pathways.

Question 5 [addressed to Qinghua Zhang, NNSA]: The CAP1400 is a scale-up of the AP1000, which is a scale-up of the AP600. The T/H behavior of the AP1000 was never verified with a full-scale mockup. Are any of the experiments you mentioned for the CAP1400 performed on a full-scale mockup?

Answer 5 [response from Qinghua Zhang, NNSA]: Independent experimental research on critical systems for CAP1400 include three subjects:

No.1: Test for residual heat removal capability of Passive core cooling system under station blockout condition and line rupture before/after PRHR isolation valve.

No.2: Test for heat removal capability of passive containment cooling system.

No.3: Test for impact of In-Vessel Retention channel change on reactor vessel heat transfer characteristic.

The experiment 1 and 2 are scheduled to be carried out on test facilities based on a Hierarchical Two-Tiered Scaling method instead of on a full-scale mockup, and the experiment 3 is scheduled to be carried out on test facilities based on a two-dimensional full-scale slice type test section.

Question 6 [addressed to Qinghua Zhang, NNSA]: Can you share with us your R&D activities in high temperature gas-cooled reactor technology?

Answer 6 [response from Qinghua Zhang, NNSA]: Tsinghua University is the designer of HTGR in China. National Nuclear Safety Administration(NNSA) issued construction permit in

2012 for HTGR nuclear power plant. The R&D activities on HTGR in Nuclear and radiation Safety Center(NSC) are regulatory oriented, including: the research and development of review principles, the content and format of safety analysis report, the in-service inspection regulatory requirement for the steam generator, the completeness of commissioning test items, and technical specifications.

Question 7 [addressed to Qinghua Zhang, NNSA]: You mentioned a Thorium Molten Salt Reactor. What is the power level of the Molten Salt Reactor? What is the intended use of the byproducts of the Thorium fuel cycle? How are the ingrowth base issues of the byproducts being managed?

Answer 7 [response from Qinghua Zhang, NNSA]: The Molten Salt Reactor(MSR) project in China will be designed as an experiment reactor with a power level of 2MW or 10MW. The designer of MSR Experiment is Chinese Academy of Sciences. The latest progress about the project is that the fuel cycle will be designed based on uranium instead of thorium in near term.

TECHNICAL SESSIONS
Thursday, March 12, 2015, 10:30 a.m. – 12:00 p.m.

TH33 Cumulative Effects of Regulation and Risk Prioritization Initiative: Operating Reactor Perspective

Session Chair: Aby Mohseni, Deputy Director, Division of Policy and Rulemaking, NRR/NRC, 301-415-6686, Aby.Mohseni@nrc.gov

Session Co-Coordinators: Jason Carneal, Project Manager, Division of Policy and Rulemaking, NRR/NRC, 301-415-1451, Jason.Carneal@nrc.gov

Steve Ruffin, Project Manager, Division of Policy and Rulemaking, NRR/NRC, 301-415-1985, Steve.Ruffin@nrc.gov

The questions below were not answered during the above session.

Question 1: Do you envision any possible nexus between CER/RPI and Project AIM? For example, could an RPI-like process be used to help prioritize what NRC staff focuses on first?

Answer 1: At this time (prior to Commission decision on Project Aim), the staff has not evaluated CER/RPI for a relationship to Project Aim. The staff is currently developing recommendations to the Commission for potential expansion of the consideration of the cumulative effects of regulation to regulatory actions other than rulemaking. As described in the staff's pre-decisional draft paper presented to the Advisory Committee for Reactor Safeguards (ACRS), available in ADAMS at accession number ML15036A181, the staff's recommended Option 2 includes a recommendation to explore piloting of an NRC expert panel that would use risk insights and other relevant technical information to make recommendations to prioritize proposed regulatory actions across the Operating Reactor business line. This information could be used by the Office of Nuclear Reactor Regulation (NRR) Office Director to ensure that the NRC's resources and skill sets are focused on the items of highest risk significance.

Question 2: Cost benefit analyses still use guidance from the 1990s where the conversion factor for risk avoidance is $2,000 per person-rem. Studies show this is now low by a factor of 2, yet cost-benefit studies continue to underestimate the risk avoidance cost. Why has this guidance not been updated?

Answer 2: The staff is in the process of updating the dollar/person-rem conversion factor. In 2010, the NRC staff began conducting research and outreach to Federal agencies on their process for implementing the value of a statistical life for use in updating the dollar per person-rem conversion factor. As discussed in SECY-12-0110, "Consideration of Economic

Consequences within the U.S. Nuclear Regulatory Commission's Regulatory Framework," the staff recommended updating numerous guidance documents, including NUREG-1530, "Reassessment of NRC's Dollar per Person-Rem Conversion Factor Policy," which is available at Agencywide Document Access and Management System (ADAMS) Accessions No. ML063470485. In the Staff Requirements Memorandum (SRM) to SECY 12-0110, the Commission approved the NRC staff's recommendations (ADAMS Accession No. ML13079A055). A public meeting is scheduled for April 2, 2015 (ADAMS Accession No. ML15068A046) to provide an overview of background and history of this conversion factor, to discuss changes between the original NUREG-1530 published in 1995 and this revision, and to discuss the schedule for publishing Revision 1 to NUREG-1530 for public comment. The staff expects to publish a draft update to NUREG-1530 for comment in later 2015.

Question 3: How is CER/RPI used in concert with an adequate protection rule?

Answer 3: Resolution of adequate protection issues takes priority over cumulative effects of regulations (CER) concerns. The NRC staff's CER efforts examine ways in which the agency may be able to enhance the efficiency with which it implements regulatory actions, while mitigating the cumulative impact of regulatory activities.

Question 4: Do you think if RPI and CER were applied in 2010 at Fukushima plant, they would have avoided the accident?

Answer 4: The NRC did not evaluate the impact CER enhancements would have had to avoid the accident at Fukushima. In its development of possible actions to address lessons learned from Fukushima, the NRC prioritized its actions to ensure the timely implementation of the most important safety improvements. The development of all post-Fukushima regulatory actions has included extensive public interaction. Furthermore, the NRC is developing associated guidance to be available during the implementation of the requirements. The NRC has also recognized the potential overlap of certain activities and is employing its CER processes as part of the rulemakings stemming from Fukushima to mitigate the cumulative impacts of the regulatory actions.

Question 5: What role does NRC enforcement of regulations play or not play in the cumulative effects of regulation?

Answer 5: The overall goal of this process, if approved by the Commission and implemented, would be to focus the licensees' attention and resources on the regulatory actions of highest safety significance for operating reactors, on a plant-specific basis. The NRC will continue to provide enforcement and oversight of licensees' activities with regards to its rules and regulations to ensure public health and safety.

Question 6: How will the process handle slicing up mods to reduce risk of big projects and ignoring the cumulative risk of the overall modification?

Answer 6: As part of the demonstration pilots, NRC staff and licensees observed the impact of separating issues into different components and documented it in the summary of the NRC staff's observations of the demonstration pilots. The summary is available at ADAMS Accession No. ML14302A269. Specifically, the NRC staff and licensees learned potential advantages and disadvantages associated with splitting a project into multiple parts. A potential disadvantage is the dilution of the overall significance of the issue. A potential advantage is the ability to determine the relative risk significance of different components of the overall project, and therefore gaining the ability to prioritize each component within the overall project schedule. NRC staff plans to continue to engage industry to address this issue in a manner that enables licensees to focus their time, attention, and resources on issues of highest safety significance.

Question 7: Can you share some of your observations from the pilot plant integrated decision making panel (IDP) and aggregation meetings from your perspective?

Answer 7: NRC staff documented observations of the demonstration pilots of the draft Nuclear Energy Institute guidance in a report entitled "Summary of Staff's Observation of Industry Demonstration Pilot Activities of NEI Draft Guidance for Prioritization and Scheduling Implementation," which is available at ADAMS Accession No. ML14302A269.

Question 8 [addressed to Mike Glover, Duke Energy]: Was the NFPA 805 risk calculation in today's terms or did it consider when the modification will be installed? Were the NFPA 805 modifications considered as one modification or multiple modifications?

Answer 8 [response from Mike Glover, Duke Energy]: The NFPA805 mods were considered separately as to their risk impact. The risk calculation was in today's terms.

Question 9 [addressed to Mike Glover, Duke Energy]: Do you think if RPI and CER were applied in 2010 at the Fukushima plant, they would have avoided the accident?

Answer 9 [response from Mike Glover, Duke Energy]: I do not think so. In my view, they missed the fundamental issue of how high of a tsunami to build into their design of the plant. Had that been properly considered, the event could have been prevented.

Question 10 [addressed to Mike Glover, Duke Energy]: Can you quantify the reduction / impact on cumulative burden? Also, can you quantify the sponsorship budget impact to O&M capital (i.e., increase or decrease)?

Answer 10 [response from Mike Glover, Duke Energy]: From our review of the site proposed modifications we cancelled 3 mods that would have cost on the order of $7 million. On the regulatory side, our decision to change a commitment on venting of ECCS piping from a monthly frequency and continuing to conduct it quarterly will have a dose reduction impact of 200 millirem per quarter.

Question 11 [addressed to Mike Glover, Duke Energy]: Are there any examples of increasing capital expenditures because of cumulative effect? If there are some examples, we can understand the impact better.

Answer 11 [response from Mike Glover, Duke Energy]: Yes- Fukushima costs are on the order of $60 million for Robinson. The NFPA 805 mods will cost over $10 million. The mods to address a single phase failure will cost around $2 million.

Question 12 [addressed to Mike Glover, Duke Energy]: Relating to GSI-191, if after all of the analyses and testing, the fibrous insulation is still not removed, how is safety improved in the plant?

Answer 12 [response from Mike Glover, Duke Energy]: It is improved by knowing where fibrous insulation is located and ensuring it is not impacted beyond the capability of the containment sump to address it in a LOCA or steam line break event due to zone of influence.

Question 13 [addressed to John Butler, NEI]: How will the process handle slicing up mods to reduce risk of big projects and ignoring the cumulative risk of the overall modification?

Answer 13 [response from John Butler, NEI]: Section 2.2.1 of NEI 14-10 provides some guidance to address this issue. Item 6 in particular says to consider the overall impact of the issue when considering other issues. Additionally, the training sessions that were given highlighted the need to take a comprehensive look. It was our experience during the pilot that as a first cut, licensees looked holistically at the issue (e.g. NFPA 805), then only afterwards, considered narrower mods.

Question 14 [addressed to John Butler, NEI]: Do you think if RPI and CER were applied in 2010 at the Fukushima plant, they would have avoided the accident?

Answer 14 [response from John Butler, NEI]: It is speculative to state with certainty that that would be the case. Certainly, if there was a regulatory issue to address tsunami or other external events, the RPI process lends itself to a logical, consistent, and repeatable approach to prioritize the issue.

Question 15 [addressed to John Butler, NEI]: Does the industry intend to strengthen PRA tools (quality and scope) to support RPI?

Answer 15 [response from John Butler, NEI]: Improvement of PRA tools and models is an continual process that is driven primarily by an ever expanding role of PRA in nuclear plant operations.

Question 16 [addressed to John Butler, NEI]: Would industry want the RPI process to identify mods that could be delayed and be a justification for a license amendment to delay a license condition modification timing (for example, NFPA 805)?

Answer 16 [response from John Butler, NEI]: The prioritization process is focused on identifying the relative importance of projects so that finite resources can be deployed more effectively.

Question 17 [addressed to John Butler, NEI]: How would a plant take into account the delay in implementing a modification or change that would result from risk prioritization? For example, available resources could result in delaying the implementation.

Answer 17 [response from John Butler, NEI]: NEI 14-10 is very explicit in how to implement the various levels of priority. For example, sufficient resources (financial and skilled personnel) should be dedicated to Priority 1 activities such that the activity will be worked with the maximum feasible effort, and so forth. Under this approach, highest priority activities would not be delayed more than they otherwise would have been without the prioritization process, and the pilot identified several instances where the highest ranking projects would have been implemented sooner.

Question 18 [addressed to John Butler, NEI]: Where are compensatory measures and operator manual actions included in the risk analyses for RPI?

Answer 18 [response from John Butler, NEI]: Section 5.1 of NEI 14-10 discusses other considerations such as impact on operator burden. Since the methodology is intended to address the long-term implementation of regulatory and plant issues, short-duration compensatory measures are not generally addressed by this approach.

Question 19 [addressed to John Butler, NEI]: Does NEI envision that the NRC would review/approve the reprioritization (site-specific) process result? Does NEI envision backstops?

Answer 19 [response from John Butler, NEI]: The reprioritization process, as demonstrated during the pilots, can identify actions to modify the schedule for plant initiated actions or actions that are driven by regulatory requirement or commitment. Any schedule changes for actions driven by regulation or regulatory commitment would be addressed using established processes. These include exemption requests per 10 CFR 50.12 or 52.7, as applicable; or following a commitment change process such as that described in NEI 99-04, Rev. 0, *Guidelines for Managing NRC Commitment Changes*.

Current guidance provides a backstop to prevent continual deferrals of low priority actions. Prioritization guidance provided in Section 5 of NEI 14-10, Revision 0, states that if an activity continues to be subject to deferral, after deferring to the third operating cycle, licensees should decide whether to begin implementation by the end of the next planned refueling outage or submit a request, using the appropriate licensing process, to eliminate the action.

Question 20 [addressed to John Butler, NEI]: Using site-specific expert panels to evaluate risks makes sense – but can licensees maintain adequate levels of independence?

Answer 20 [response from John Butler, NEI]: The pilot at six reactor sites clearly showed that the plant-specific multi-disciplinary IDPs brought to the table wide-ranging perspectives. IDP panels already are used for a multitude of risk-informed initiatives including Maintenance Rule and 50.69.

Question 21 [addressed to John Butler, NEI]: "Cumulative Effects of Regulation" is framed as a problem. Can it also be beneficial, when regulations cover each other's blind spots as a kind of defense in depth?

Answer 21 [response from John Butler, NEI]: Impact on defense in depth is a consideration in the NEI 14-10 approach. However, keep in mind that the prioritization process is focused on identifying the relative importance of projects so that finite resources can be deployed more effectively.

Question 22 [addressed to John Butler, NEI]: Significance Determination Process (SDP) thresholds intended for punitive actions assign a "high" (red) priority only on a rare basis (>10 in 10,000 year chance of core damage). With such a high threshold, how often, if ever, would a plant initiative trigger a "high" priority? Why not adopt thresholds over, for example, RG 1.174, that allow more discrimination among plant issues at more typically encountered levels?

Answer 22 [response from John Butler, NEI]: The pilots identified two Priority 1 issues out of about 105 evaluated. This is as expected. Since prioritization is relative, the absolute number of Priority 1, 2, 3, and so forth is not as important as the relative ranking of the projects and issues. NEI 14-10 provides additional guidance for discriminating within a Priority level based on additional considerations such as impact on operator burden.

TH34 Long-Term Performance of Cast Austenitic Stainless Steel Reactor Internal Components

Session Chair: Robert Tregoning, Senior Technical Advisor for Materials Engineering Issues, Division of Engineering, RES/NRC, 301-251-7662, Robert.Tregoning@nrc.gov

Session Co-Coordinators: Makuteswara Srinivasan, Senior Materials Engineer, Division of Engineering, RES/NRC, 301-251-7630, Makuteswara.Srinivasan@nrc.gov

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

TH35 Safety Assurance in Digital Safety Systems

Session Chair: Sushil Birla, Senior Technical Advisor, Division of Engineering, RES/NRC, 301-251-7660, Sushil.Birla@nrc.gov

Session Coordinator: Bernard Dittman, Digital I&C Engineer, Division of Engineering, RES/NRC, 301-251-7494, Bernard.Dittman@nrc.gov

The questions below were not answered during the above session.

Question 1 [addressed to Sofia Guerra, Adelard LLP]: Using a basis of reasoning with a body of evidence, even though convincing (in whose mind), is still subjective. This provides uncertainty in the regulatory environment. How do you drive this type of process to a high confidence assessment (a number) for cost/benefit analysis? Interpretation = uncertainty => increased costs.

Answer 1 [response from Sofia Guerra, Adelard LLP]: I agree with Tim's Answer. Although (detailed) prescriptive approaches might seem to reduce subjectivity, this is not the case. There is subjectivity not only in the interpretation of the requirements, but also in whether the requirements have been adequately met. I've worked on multiple compliance cases as well as assessing cases done by others, and it is obvious the level of interpretation that is required, which needs to take into account the specific characteristics of the systems being considered. Assurance cases give you a mechanism of communicating such reasoning, and therefore reducing uncertainty in the interpretation => reducing costs.

Answer 1 [response from Tim Kelly, University of York, United Kingdom]: My first response is that the subjectivity isn't *created* by the use of an assurance case. The subjectivity (e.g. in the interpretation of a requirement in a standard, or in the justification of the adequacy of hazard analysis) was / is always there. A key virtue in assurance cases is that it makes this subjectivity *transparent*. It becomes exposed and can therefore be subject to review. For example I've been involved in projects where the safety case has been poor / non-existent and in the end a major sticking point has been the justification of the Safety Integrity Level (Risk Classification) that was set for the system at a very early stage. If, as best practice would suggest, such justifications and arguments were established, explicitly presented, and reviewed in step with the project development, there would have been an opportunity to manage this (project) risk. No explicit safety argument + subjectivity, bias, interpretation = undisclosed, and therefore unmanaged, risks => increased costs.

Answer 1 [response from Sushil Birla, NRC]: Referring to the second paragraph of the Question, the scope of TH-35 did not include either "confidence assessment (a number)" or "cost/benefit analysis." Referring to the premise in the first sentence of the Question, "using a basis of reasoning with a body of evidence is still subjective," the implied subjective interpretation is reduced through explicit reasoning as follows: (1) Formulate and state the reasoning in terms of logical propositions. Each logical proposition should make the reasoning clear (i.e., the assertion should follow directly from the premise). The premise in a statement may be the assertion in a supporting proposition. This creates a chain of propositions that should end with evidence grounded in some repeatable measurement (e.g., as verification that a unit of software in the system satisfies the requirements allocated to it). (2) Clearly relate the verification specifications (e.g., including test cases) to the requirements whose satisfaction is being verified (i.e., demonstrated through propositional statements mentioned above). (3) Show (i.e., demonstrate through propositional statements mentioned above) that the requirements control (i.e., eliminate; avoid; mitigate) all hazards identifies in hazard analysis. (4) At every level of integration of the system, show (i.e., demonstrate through propositional statements mentioned above) that the requirements for that level satisfy the requirements at the next higher level of integration and do not introduce any other behavior. (5) At every level of integration of the system, show (i.e., demonstrate through propositional statements mentioned above) that the requirements for that level are verifiable and repeatable by different parties. (6) Identify explicitly any deficit in the preceding steps. (7) Provide the reasoning to substantiate that the deficit is not safety significant (i.e., the deficit does not lead to degradation of a safety function).

Question 2 [addressed to Darren Cofer, Rockwell Collins Advanced Technology Center):
In the aerospace industry are the modelling/analytical tools used to test the SW considered as independent verification & validation (IV&V)? Also, how do you validate these tools?

Answer 2 [response from Darren Cofer, Rockwell Collins Advanced Technology Center):
Independent verification and validation (IV&V) is normally taken to mean V&V performed by a third-party organization not involved in the development of the product. DO-178C requires that some (but not all) verification activities for Level A & B software (the most critical) be performed by someone other than the software developer. A tool may be used to achieve independence. All software QA activities must be performed with independence, including the authority to ensure corrective action.

In commercial aerospace certification, tool qualification (not validation) is the process necessary to obtain certification credit for the use of a tool. Qualification of a tool is needed when certification processes are eliminated, reduced, or automated by the use of a software tool without its output being verified. DO-330 provides guidance for tool qualification.

Question 3 [addressed to Darren Cofer, Rockwell Collins Advanced Technology Center):
What are examples of new technologies to meet critical software [*sic*]? What is an example of a mathematical technique to demonstrate that software is compliant? Postulate how a mathematical technique would be used to demonstrate how software would meet NRC regulations/requirements for safety critical systems.

Answer 3 [response from Darren Cofer, Rockwell Collins Advanced Technology Center]:
Perhaps you are asking about the bullet on one of my slides that says “New technologies that challenge the existing certification process.” One example of this is Model-Based Development (MBD) for software (tools like Simulink and SCADE) which facilitates design at a higher level of abstraction, simulation, and subsequent automatic generation of source code. MBD has been widely adopted because of productivity gains (at least in some applications) but poses

challenges to the existing certification process. DO-331 attempts to address some of these challenges.

1. Theorem proving, model checking, and abstract interpretation are three categories of formal analysis techniques that have been applied to software verification. For examples of all three applied to avionics software, see the case studies described here: http://loonwerks.com/projects/do333.html.

2. The claim is that mathematical techniques may be used to verify that software, software models, or system models satisfy their requirements, not the overarching NRC regulations. The case studies mentioned in (2) provide examples of what this looks like in the commercial aircraft domain.

Answer 3 [response from Sofia Guerra, Adelard LLP]: Regarding Darren's response to point (2), I would like to add that those techniques have been used in the nuclear industry in several countries in Europe as well. While not intending to be comprehensive, here are examples where mathematical techniques have been used in the nuclear industry in Europe.

Abstraction interpretation and theorem proving were applied to the Sizewell B PPS. This is described in D Pavey, L Winsborrow, Formal demonstration of equivalence of source code and PROM contents: an industrial example, Nuclear Systems Branch, Nuclear Electric, In Chris Mitchell and Victoria Stavridou, editors, “Mathematics of Dependable Systems”, pages 225–248. Clarendon Press, 1995.

"Assessment and Qualification of Smart Sensors" S Guerra, P Bishop, R Bloomfield, D Sheridan In Proceedings NPIC/HMIT 2010, Las Vegas, USA, 2010, also talks about abstract interpretation and theorem proving for smart sensors in the United Kingdom.

The following two papers applied abstract interpretation to a nuclear protection system outside of the United Kingdom:

- "Software Criticality Analysis of COTS/SOUP" Peter Bishop, Robin Bloomfield, Tim Clement, Sofia Guerra. Reliability Engineering and System Safety 81 (2003) 291-301.
- "Integrity Static Analysis of COTS/SOUP" P.G. Bishop, R.E. Bloomfield, T.P. Clement, A.S.L. Guerra and C.C.M. Jones. In Proceedings SAFECOMP 2003, pp. 63-76, 21-25 Sep, Edinburgh, UK, 2003, (c) Springer Verlag

Model checking for Finnish nuclear power plants is discussed in the following references:

- http://www.iaea.org/NuclearPower/Downloadable/Meetings/2011/2011-05-24-05-26-TWG-NPPIC/Day-1.Tuesday/FINLAND_NPPIC_STUK_HH.pdf.
- "Model checking of I&C software in the Loviisa NPP automation renewal project" Pakonen, Antti; Valkonen, Janne; Matinaho, Sami; Hartikainen, Markus. Citation Automaatio XXI, 17 - 18.3.2015, Helsinki, Finland. Suomen Automaatioseuran

julkaisusarja nro 44, ISBN 13 978–952-5183-46-7, Date 2015, Rights Finnish Society of Automation. This article may be downloaded for personal use only.

- "Model Checking for Licensing Support in the Finnish Nuclear Industry" Pakonen, Antti; Valkonen, Janne; Matinaho, S; Hartikainen, M. Citation International Symposium on Future I&C for Nuclear Power Plants (ISOFIC 2014), Jeju Island, Republic of Korea, 24 - 28 August 2014, Date 2014. This article may be downloaded for personal use only.

Question 4 [addressed to Darren Cofer, Rockwell Collins Advanced Technology Center): How is the aviation industry dealing with the immense number of uncertainties associated with the supply chain of COTS components?

Answer 4 [response from Darren Cofer, Rockwell Collins Advanced Technology Center): I am not an expert in the use of COTS hardware in aircraft. In general, all onboard systems (radios, displays, flight management system, etc.) are built specifically for that application. However, the computing components (CPUs, memory, etc.) are COTS. Very little onboard software is COTS (RTOS software is one example), but even that is required by DO-178C to satisfy most of the same certification objectives as custom software. Section 12.3.4 of DO-178C makes some provision for the use of service history in establishing the suitability of COTS software.

Question 5 [addressed to Darren Cofer, Rockwell Collins Advanced Technology Center): Can you please give examples of defects common in requirements phase? (top line of last slide)

Answer 5 [response from Darren Cofer, Rockwell Collins Advanced Technology Center]: Examples of defects in the requirements phase include:

- Missing or incomplete requirements
- Imprecise or ambiguous requirements
- Unstated assumptions
- Unstated constraints

Additionally, it is often the case that human errors introduce defects when moving from high level to lower level requirements, or in producing a design that correctly implements low level requirements. Formalization of requirements and the use of analysis tools help to discover these kinds of defects.

Answer 5 [response from Sushil Birla, NRC]: In the Answer by Dr. Cofer, the expression, “defects when moving from high level to lower level requirements,” refers to the process of decomposition and derivation of requirements, in which the originally specified behavior may not be satisfied or unspecified behavior, may be introduced. Such defects may be prevented through the process of strict stepwise refinement, discussed in Appendix D of Research Information Letter 1101,”Technical basis to review hazard analysis of digital safety systems,” NRC Agencywide Documents Access Management System (ADAMS) Accession No. ML14237A359.

TH36 Future Direction of International Research for Reactors and Fuel Cycle Safety (Part 2)

Session Chair: Brett Rini, International Programs Team Leader (Acting), RES/NRC, 301-251-7615, Brett.Rini@nrc.gov

Session Coordinator: Wendy Eisenberg, Research Program Assistant, RES/NRC, 301-215-7682, Wendy.Eisenberg@nrc.gov

The questions below were not answered during the above session.

Question 1 [addressed to Dr. Frank-Peter Weiss, GRS]: Has the loss of nuclear power adversely impacted non-power nuclear activities such as nuclear medicine?

Answer 1 [response from Dr. Frank-Peter Weiss, GRS]: The German phase-out decision exclusively relates to NPPs. Non-power nuclear activities have not been impacted until now. No research reactor was shut down. Production of radio-isotopes for medical purposes even enhanced. High energy ion therapy of cancer diseases is offered by an increasing number of hospitals. Nevertheless, future adverse influences cannot be excluded.

Question 2 [addressed to Dr. Frank-Peter Weiss, GRS]: Are there still young adults interested in studying anything nuclear at the Universities?

Answer 2 [response from Dr. Frank-Peter Weiss, GRS]: The number of students interested in a specialization in nuclear technology went significantly down in spite a few universities (Dresden, Munich, Karlsruhe, Jülich, Bochum) still offer Master and Diploma courses in nuclear technology and nuclear energy. However, we (GRS) can still hire a sufficient number of highly qualified young experts from a large variety of disciplines.

Question 3 [addressed to Jean-Claude Micaelli, IRSN]: What is the schedule for the ASTRID reactor?

Answer 3 [response from Jean-Claude Micaelli, IRSN]: The pre-conceptual design phase should be completed by the end of 2015; it should be followed up to 2019 by a detailed design phase which should allow a commissioning during around 2025.

Question 4 (addressed to Jean-Claude Micaelli, IRSN): Is there any chance that an international collaboration, like what was done for fusion (ITER) could be assembled for fast reactor development or, have fast reactor designs matured enough (e.g., Russian BN300 reactors) such that international collaboration is not necessary?

Answer 4 (answered by Jean-Claude Micaelli, IRSN): ASTRID reactor cannot be compared to ITER. ASTRID is a demonstrator that aims to validate the technical options of GEN IV commercial SFRs that could be deployed (according to CEA: designer and future operator of ASTRID) beyond 2040.

ITER is a research machine that aims to demonstrate the principle of producing more energy from fusion than used to create and keep stable fusion plasma. We are not yet at the stage of a fusion power plant demonstrator.

An international collaboration has been put in place around the ASTRID project, it is nevertheless not as large as for ITER; ASTRID project is today mainly funded by France, in the perspective of a possible deployment, in particular in France, of GEN IV SFRs in the second half of the century.

TH37 Unique Aspects of Regulating Research and Test Reactors

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

Session Coordinator: Cindy Montgomery, Project Manager, Division of Policy and Rulemaking, NRR/NRC, 301-415-3398, Cindy.Montgomery@nrc.gov

The questions below were not answered during the above session.

Question 1 [addressed to Alexander Adams, Jr., NRC]: Would your organization be well-equipped to license a prototype reactor using non-water technology (Gen 4 molten salt) or would that effort be better served by working with NRO?

Answer 1 [response from Alexander Adams, Jr., NRC]: The Research and Test Reactors Licensing Branch (PRLB) within the Office of Nuclear Reactor Regulation have responsibility for licensing non-power reactors, including first-of-a-kind designs. The Office of New Reactors also could have responsibility for the licensing of new designs. Both offices work closely together in this area and would determine a licensing path based on the specifics of the licensing request. The NRC staff encourages potential applicants proposing novel technologies to engage with the NRC early in the design process to facilitate efficient licensing reviews.

Question 2 [addressed to Alexander Adams, Jr., NRC]: What is the next generation of research reactors that the NRC regulates going to look like? (e.g. SMR, fusion, Gen IV)

Answer 2 [response from Alexander Adams, Jr., NRC]: The next generation of non-power reactors to be regulated by the NRC is dependent on the licensing requests received from applicants. For example, the Research and Test Reactors Licensing Branch (PRLB) within the Office of Nuclear Reactor Regulation is currently reviewing construction permit applications for non-power facilities proposing to produce medical radioisotopes. The PRLB staff has also met with potential applicants proposing non-power molten salt reactor technologies.

Question 3 [addressed to Alexander Adams, Jr., NRC]: This is slightly off the topic of your talk, but could you address the Atomic Energy Act and/or the NRC's regulations to permit foreign ownership of research reactors, to possibly encourage new RTR deployment in the U.S.?

Answer 3 [response from Alexander Adams, Jr., NRC]: At the present time, indirect foreign ownership of up to, but not including, 100% is permitted. However, at this time there are no indirect foreign owners of more than 50% of any facility licensed under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50. By indirect ownership, we mean the parent company of the licensee. Direct foreign ownership of reactors is not permitted under the AEA, sections 103d. and 104d. Section 104d. is relevant to RTRs.

Question 4 [addressed to Jeff Chamberlin, DOE]: Does the January 2016 deadline for a uranium lease/take-back program look achievable?

Answer 4 [response from Jeff Chamberlin, DOE]: Yes, and the Department of Energy is providing regular updates on development of the program at Uranium Lease and Take Back program (ULTB) sessions on the margins of the periodic Office of Science and Technology Policy (OSTP)-led Mo-99 Stakeholders Meetings.

Question 5 [addressed to Jeff Chamberlin, DOE]: What numbers of the 52 HEU facilities were converted (instead of shutdown). What types of fuels were used? How were their power and operating cycles impacted.

Answer 5 [response from Jeff Chamberlin, DOE]: To date a total of 92 research reactors and isotope production facilities have either been converted or verified as shutdown, 52 of which were completed since the Global Threat Reduction Initiative started in 2004. 27 of those 52 reactors were converted from the use of high enriched uranium (HEU) fuel to low enriched uranium (LEU) fuel. Throughout this effort the Convert Program has used multiple fuel types in the conversion process, including 12 with silicide fuel, 5 with TRIGA fuel, and one solution reactor. The remaining 9 reactors were converted with oxide fuel.

The policy of the Department of Energy Convert Program has been to maintain a performance loss of less than 10% after conversion. The program works with each facility to define performance loss by assessing key performance metrics for individual reactor mission needs. Each reactor's unique mission and performance requirements vary. For some reactors, power and fuel cycles are key performance requirements, while other reactors may have different distinct factors that are more important. Despite these ever changing objectives, through 66 conversions to date, the program has consistently accomplished its objective to convert reactors within the 10% performance loss margin.

Question 6 [addressed to Jeff Chamberlin, DOE]: Most facilities using HEU are quite old (40+ years). Is it justified to spend all those efforts in conversion or should one go for new facilities, respecting today's safety requirements?

Answer 6 [response from Jeff Chamberlin, DOE]: Efforts of the Department of Energy Convert Program are intended to support the acceleration of civilian high enriched uranium (HEU) minimization. While the United States is not in a position to decide for other states what they should do with their reactors, the Convert Program has found through extensive experience converting reactors that providing support for conversion to low enriched uranium (LEU) creates a greater likelihood for open cooperation and successful HEU minimization when compared to pressing reactor stakeholders for shutdown and possible replacement. Moreover, the costs of replacing reactor facilities are prohibitive. In fact, a DOE working group was stood up in 2013 to look at this very Question with regard to the United States' nuclear research complex and the group found that the existing facilities will operate well into the future and are currently the optimal option for meeting irradiation needs of the future. Therefore, reactor conversion efforts should continue to address the remaining U.S. HEU-fueled reactors.

Question 7 [addressed to Joe Staudenmeier, NRC]: What is your guess of the impact of improved TRACE and RELAP5 on the source term of TRIGA (severe accident)?

Answer 7 [response from Joe Staudenmeier, NRC]: TRACE and RELAP5 are not used to perform source term calculations in severe accidents. These are used to evaluate the thermal margin of the reactor fuel to help insure the integrity of the first barrier to fission product release.

Question 8 [addressed to Joe Staudenmeier, NRC]: Has there been any interest in using the thermal hydraulic codes for confirmatory analyses at the Advance Test Reactor (ATR) or soon to be restarted TREAT test reactor at INL? Or any SNL reactors?

Answer 8 [response from Joe Staudenmeier, NRC]: The Department of Energy (DOE) uses thermal hydraulic codes to evaluate the thermal margin in their research and test reactors but DOE is not under NRC regulation.

Question 9 [addressed to Ralph Butler, University of Missouri Research Reactor]: How does the NRC regulate Nuclear Batteries?

Answer 9 [response from Ralph Butler, University of Missouri Research Reactor]: Nuclear batteries are not yet commercially available and thus not regulated.