U.S.NRC

United States Nuclear Regulatory Commission

Protecting People and the Environment

NUREG-1650, Rev. 5
Supplement 1

Answers to Questions from the Peer Review by Contracting Parties on the United States of America Sixth National Report for the Convention on Nuclear Safety

Office of Nuclear Reactor Regulation

AVAILABILITY OF REFERENCE MATERIALS IN NRC PUBLICATIONS

NRC Reference Material

As of November 1999, you may electronically access NUREG-series publications and other NRC records at NRC's Public Electronic Reading Room at http://www.nrc.gov/reading-rm.html. Publicly released records include, to name a few, NUREG-series publications; *Federal Register* notices; applicant, licensee, and vendor documents and correspondence; NRC correspondence and internal memoranda; bulletins and information notices; inspection and investigative reports; licensee event reports; and Commission papers and their attachments.

NRC publications in the NUREG series, NRC regulations, and Title 10, "Energy," in the *Code of Federal Regulations* may also be purchased from one of these two sources.

1. The Superintendent of Documents
   U.S. Government Printing Office
   Mail Stop SSOP
   Washington, DC 20402–0001
   Internet: bookstore.gpo.gov
   Telephone: 202-512-1800
   Fax: 202-512-2250
2. The National Technical Information Service
   Springfield, VA 22161–0002
   www.ntis.gov
   1–800–553–6847 or, locally, 703–605–6000

A single copy of each NRC draft report for comment is available free, to the extent of supply, upon written request as follows:

Address: U.S. Nuclear Regulatory Commission
Office of Administration
Publications Branch
Washington, DC 20555-0001

E-mail: DISTRIBUTION.RESOURCE@NRC.GOV
Facsimile: 301–415–2289

Some publications in the NUREG series that are posted at NRC's Web site address http://www.nrc.gov/reading-rm/doc-collections/nuregs are updated periodically and may differ from the last printed version. Although references to material found on a Web site bear the date the material was accessed, the material available on the date cited may subsequently be removed from the site.

Non-NRC Reference Material

Documents available from public and special technical libraries include all open literature items, such as books, journal articles, transactions, *Federal Register* notices, Federal and State legislation, and congressional reports. Such documents as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings may be purchased from their sponsoring organization.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at—

The NRC Technical Library
Two White Flint North
11545 Rockville Pike
Rockville, MD 20852–2738

These standards are available in the library for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from—

American National Standards Institute
11 West 42nd Street
New York, NY 10036–8002
www.ansi.org
212–642–4900

Legally binding regulatory requirements are stated only in laws; NRC regulations; licenses, including technical specifications; or orders, not in NUREG-series publications. The views expressed in contractor-prepared publications in this series are not necessarily those of the NRC.

The NUREG series comprises (1) technical and administrative reports and books prepared by the staff (NUREG–XXXX) or agency contractors (NUREG/CR–XXXX), (2) proceedings of conferences (NUREG/CP–XXXX), (3) reports resulting from international agreements (NUREG/IA–XXXX), (4) brochures (NUREG/BR–XXXX), and (5) compilations of legal decisions and orders of the Commission and Atomic and Safety Licensing Boards and of Directors' decisions under Section 2.206 of NRC's regulations (NUREG–0750).

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

NUREG-1650, Rev. 5
Supplement 1

Answers to Questions from the Peer Review by Contracting Parties on the United States of America Sixth National Report for the Convention on Nuclear Safety

Manuscript Completed: March 2014
Date Published: November 2014

Prepared by
U.S. Nuclear Regulatory Commission (NRC)
Institute of Nuclear Power Operations (INPO)

Office of Nuclear Reactor Regulation

ABSTRACT

The Convention on Nuclear Safety (CNS) was adopted in June 1994 and entered into force in October 1996. The objectives of the CNS are to achieve and maintain a high level of nuclear safety worldwide. Contracting parties to the Convention have four obligations: submit a national report for peer review, review the national reports of other contracting parties, respond to questions and comments submitted by the contracting parties, and participate in the organizational and review meetings. The United States published its sixth national report for peer review in October 2013 (NUREG-1650, "The United States of America National Report for the Convention on Nuclear Safety: Sixth National Report, October 2013," Revision 5). Addendum 5 to NUREG-1650 documents the answers to questions raised by contracting parties during their peer reviews of the U.S. national report. Specifically, the questions and answers resulting from the peer reviews concern the safety of existing nuclear installations, the legislative and regulatory framework, the regulatory body, responsibility of the licensee holder, priority to safety, financial and human resources, human factors, quality assurance, assessment and verification of safety, radiation protection, emergency preparedness, siting, design and construction, and operation, and implementation of the lessons learned from the Fukushima accident. The sixth review meeting of the CNS was held at the International Atomic Energy Agency in Vienna, Austria, from March 24 through April 4, 2014.

TABLE OF CONTENTS

EXECUTIVE SUMMARY

The objectives of the Convention on Nuclear Safety (CNS) are to achieve and maintain a high level of nuclear safety worldwide. Contracting parties to the CNS have four obligations: submit a national report for peer review, review the national reports of other contracting parties, respond to questions and comments submitted by the contracting parties, and participate in the organizational and review meetings.

The United States published its sixth national report for peer review in October 2013 (NUREG-1650, "The United States of America National Report for the Convention on Nuclear Safety: Sixth National Report, October 2013," Revision 5), which is available on the U.S. Nuclear Regulatory Commission's (NRC's) Web site at http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1650/. Addendum 5 to NUREG-1650 documents the answers to questions raised by contracting parties during their peer reviews of the U.S. national report.

On receiving questions from contracting parties, the NRC staff categorized them according to the article of the U.S. national report that addressed the relevant material. Subsequently, technical and regulatory experts at the NRC and members of the Institute of Nuclear Power Operations answered the questions. These answers were provided to the contracting parties in preparation for the sixth review meeting of the CNS, which was held at the International Atomic Energy Agency in Vienna, Austria, from March 24 through April 4, 2014.

ACKNOWLEDGMENTS

Contributors to this report include the following technical and regulatory experts from the U.S. Nuclear Regulatory Commission:

Akstulewicz, Frank
Albert, Michelle
Albert, Ronald
Alley, David
Alvarado, Rossnyev
Anderson, Joseph
Armstrong, Kenneth
Audrain, Margaret
Bagchi, Manash
Balazik, Michael
Barczy, Theresa
Bauer, Laurel
Beall, Robert
Beasley, Benjamin
Boyce, Tom
Brochman, Philip
Burkhart, Larry
Cai, June
Carpenter, Gene
Casey, Lauren
Circle, Jeffrey
Clark, Lisa
Coleman, Nicole
Conley, Maureen
Cool, Donald A.
Cowdrey, Christian
Coyne, Kevin
Cruz, Luis
Culp, Lisa
Cumblidge, Stephen
Davis, Kristin
Decker, David
Demoss, Gary
Denning, Robert
DiFrancesco, Nick
Dinitz, Ira
Dorman, Dan
Dougherty, Jay
Dusaniwskyj, Michael
Erlanger, Craig
Feintuch, Karl
Ferrell, Kimberly
Fields, Leslie
Folk, Kevin
Frahm, Ronald
Franovich, Rani
Frumkin, Daniel
Gall, Jennifer
Garry, Steven
Gilberston, Anders
Glenn, Nicole
Gnugnoli, Giorgio
Gratton, Chris
Halter, Mandy
Hardies, Robert
Harrington, Holly
Harris, Charlie
Harrison, Donnie
Hart, Michelle
Henderson, Karen
Hill, Brittain
Hopkins, Jon
Hsu, Caroline
Huffman, William
Isom, James
Issa, Alfred
Jackson, Chris
Jefferson, Steven
Johnson, Robert
Joshi, Ravi
Kahler, Robert
Karipineni, Nageswara
Karwoski, Kenneth
Kavanagh, Kerri
Keefe, Molly
Kowalczik, Jeff
Lain, Paul
Lamb, John
Lapinski, George
Love-Blair, Angella
Lyons-Burke, Kathy
Mahoney, Michael
Markley, Christopher
Marshall, Jane
Martin, Kamishan
Masse, Todd
McGovern, Denise
Meighan, Sean
Mitchell, Matt
Mossman, Timothy
Moyer, Carol
Mrowca, Lynn
Mroz, Sara
Muller, David
Murray, Charles
Norris, Wallace
Norton, Charles
Nourbakhsh, Hossein
Nove, Carol
O'Neal, Daniel
Pasquale, Daniel
Pearson, Alayna
Pederson, Perry
Pedersen, Roger
Peterson, Gordon
Pieringer, Paul
Poole, Brooke
Poole, Justin
Quinones-Navarro, Lauren
Reckley, William
Rini, Brett
Robinson, Jay
Roche, Kevin
Rodriguez, Veronica
RoqueCruz, Carla
Rosales-Cooper, Cindy
Sakadales, Antoinette
Salisbury, Leah
Sallman, Ahsan
Santos, Cayetano
Scott, Tracy
Schaaf, Robert
Schmidt, Duane
Schwartz, Maria
Schwartzman, Jennifer
Shoop, Undine
Sigmon, Rebecca
Simmons, Anneliese
Simpson, JoAnn
Skeen, David
Smith, Shawn
Somerville, Glenda
Suttenberg, Jeremy
Swain, Patricia
Taneja, Dinesh
Tate, Travis
Temple, Jeffrey
Thomas, Eric

Thomas, George
Thompson, Catherine
Truong, Tung
Vechioli, Lucieann
Watson, Bruce
Weber, Carl
Weerakkody, Sunil
Westreich, Barry
Whited, Jeffrey
Wilkins, Lynnea
Williams, Donna
Williams, Kevin
Wong, Albert
Wong, Emma

Contributors to this report include the following experts from the Institute of Nuclear Power Operations:

Farr, David
Spinnato, Roger

ABBREVIATIONS

ACRR	annular core research reactor
ACRS	Advisory Committee on Reactor Safeguards
ADAMS	Agencywide Documents Access and Management System (NRC)
ADDIE	analysis, design, development, implementation, and evaluation
AEA	Atomic Energy Act of 1954, as amended
ALARA	as low as reasonably achievable
AMP	aging-management program
AMR	aging-management review
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASME	(formerly an abbreviation for “American Society of Mechanical Engineers”)
ASTM	(formerly an abbreviation for “American Society for Testing and Materials”)
ATR	advanced test reactor
ATR-C	advanced test reactor—critical
ATWS	anticipated transient without scram
BADGER	boron areal density gauge for evaluating racks
BAT	best available technology
BRIIE	baseline risk index for initiating events
BWR	boiling-water reactor
CFR	*Code of Federal Regulations*
CFSI	counterfeit, fraudulent, or suspect item
CFSIs	counterfeit, fraudulent, and suspect items
CNS	Convention on Nuclear Safety
COL	combined license
CP	contracting party
DC	design certification *or* direct current
DOE	U.S. Department of Energy
EA	enforcement action
ECCS	emergency core-cooling system
ENSI	Swiss Federal Nuclear Safety Inspectorate
ENSREG	European Nuclear Safety Regulators Group
EPA	U.S. Environmental Protection Agency
EPRI	Electric Power Research Institute
ESCP	Extended Storage Collaboration Program
ESRP	environmental standard review plan
FEMA	U.S. Federal Emergency Management Agency
FR	*Federal Register*
FSAR	final safety analysis report
FY	fiscal year
GALL	generic aging lessons learned
GEIS	generic environmental impact statement

GDC	general design criterion
GL	generic letter
HFIR	high flux isotope reactor
HFIS	human factors information system
I&C	instrumentation and control
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
IEEE	Institute of Electrical and Electronics Engineers
IER	INPO event report
IMC	inspection manual chapter
IN	information notice
INPO	Institute of Nuclear Power Operations
IP	inspection procedure
IPEEE	individual plant examination for external events
IRRS	integrated regulatory review service
IRS	incident reporting system
ISG	interim staff guidance
ISI	inservice inspection
ISO	International Organization for Standardization
ISOCCER	integrity, service, openness, commitment, cooperation, excellence, and respect
JLD	Japan Lessons-Learned Project Directorate (NRC)
KI	potassium iodide
LTO	long-term operation
LWR	light-water reactor
MD	management directive
mSv	millisievert(s)
NEI	Nuclear Energy Institute
NEPA	National Environmental Policy Act of 1969
NFPA	National Fire Protection Association
NIMS	National Incident Management System
NMSS	Office of Nuclear Materials Safety and Safeguards (NRC)
No.	number
NPP	nuclear power plant
NPSH	net positive suction head
NRC	U.S. Nuclear Regulatory Commission
NRO	Office of New Reactors (NRC)
NRR	Office of Nuclear Reactor Regulation (NRC)
NSIR	Office of Nuclear Security and Incident Response (NRC)
NTTF	near-term task force
OE	Office of Enforcement (NRC)
OPA	Office of Public Affairs (NRC)
OSART	Operational Safety Assessment Review Team
PEO	period of extended operation

PPE	plant-parameter envelope
PRA	probabilistic risk assessment
PRIS	power reactor information system
PSA	probabilistic safety assessment
PSR	periodic safety review
PWR	pressurized-water reactor
RASCAL	radiological assessment system for consequence analysis
RASP	risk assessment of operational events
rcry	reactor critical year
RES	Office of Nuclear Reactor Research (NRC)
RFI	request for information
RG	regulatory guide
RIC	regulatory information conference
RIS	regulatory issue summary
ROP	reactor oversight process
RPV	reactor pressure vessel
SALTO	safety aspects of long-term operation of water-moderated reactors
SAMA	severe-accident mitigation alternative
SAMGs	severe-accident management guidelines
SBO	station blackout
SCCS	safety culture and climate survey
SFP	spent fuel pool
SG	steam generator
SGTR	steam generator tube rupture
SPAR	Standardized Plant Analysis Risk
SRM	staff requirements memorandum
SSC	structure, system, or component
SSCs	structures, systems, and components
TEDE	total effective dose equivalent
TLAA	time-limited aging analysis
TREAT	transient reactor test facility
TSO	technical-support organization
U.S.	United States
UFSAR	updated final safety analysis report
UHS	ultimate heat sink
USA	United States of America
USAID	U.S. Agency for International Development
WANO	World Association of Nuclear Operators

STRUCTURE OF THE REPORT

This report documents the answers of the United States to questions raised by contracting parties to the Convention on Nuclear Safety (CNS or "the Convention") during their peer reviews of "The United States of America for the Convention on Nuclear Safety: Sixth National Report, October 2013" (NUREG-1650, Revision 5) (hereinafter referred to as the U.S. Sixth National Report). On receiving questions from contracting parties, the U.S. Nuclear Regulatory Commission (NRC) staff categorized them according to the article of the report that addressed the relevant material. Subsequently, technical and regulatory experts at the NRC and members of the Institute of Nuclear Power Operations (INPO) answered the questions. Please note that, with the exception of bracketed expansions added for some abbreviations that are not expanded in their answers, the questions are presented exactly as they were received, without being edited for grammar or spelling or in any other way. Also, the answers to the questions reflect the status from March 2014, which is when the answers were submitted to the International Atomic Energy Agency.

This report follows the format of the U.S. Sixth National Report for the CNS. Sections are numbered according to the article of the Convention under consideration. Each section begins with the text of the article, followed by an overview of the material covered by the section and the questions and answers that pertain to that section. This report begins with an introduction and continues with Articles 6 through 19. Specifically, these articles address the safety of existing nuclear installations, the legislative and regulatory framework, the regulatory body, the responsibility of the licensee, priority given to safety, financial and human resources, human factors, quality assurance, assessment and verification of safety, radiation protection, emergency preparedness, siting, design, construction, and operation. To be consistent with the U.S. Sixth National Report, this report does not contain sections for Articles 1 through 5. In accordance with Article 1 of the CNS, the U.S. Sixth National Report illustrated how the U.S. Government meets the objectives of the Convention. It discussed the safety of nuclear installations according to their definition in Article 2 and the scope of Article 3 and addressed implementing measures (such as national laws, legislation, regulations, and administrative means) according to Article 4. Lastly, the submission of the U.S. Sixth National Report fulfilled the obligation of Article 5.

This report cites a number of documents that are contained in the NRC's Agencywide Documents Access and Management System (ADAMS). ADAMS is a Web-based information system that provides access to all documents made public by the NRC since November 1, 1999. ADAMS permits full searching and the ability to view document images, download files, and print locally. ADAMS can be accessed from the NRC Web site (http://www.nrc.gov/reading-rm/adams.html). In addition, documents are available through the NRC's Public Document Room. One may contact the Public Document Room in any of the following ways:

Telephone:	1-800-397-4209 or 301-415-4737
TDD (for the hearing impaired):	1-800-635-4512
Facsimile:	301-415-3548
U.S. Mail:	U.S. NRC, PDR, O1F13, Washington, DC 20555
Onsite visit:	11555 Rockville Pike, Rockville, MD 20852
Internet:	http://www.nrc.gov/reading-rm/contact-pdr.html

INTRODUCTION TO THE U.S. SIXTH NATIONAL REPORT

This section of the U.S. Sixth National Report for the CNS described the following:

- purpose and structure of the report
- summary of changes since the previous report was written in 2010
- U.S. national policy on nuclear activities
- national nuclear programs
- safety and regulatory issues, and regulatory accomplishments
- international peer reviews and missions

The questions below were submitted by contracting parties on this section of the report.

Question Number (No.) 1
The item "Tier 1 Recommendations" (Summary; Section 1.3.2; page 31), mentions that the first tier consists of actions that could begin without unnecessary delay. Could you give more information about which are the specific "station blackout (SBO) regulatory actions - Recommendation 4.1-"?
Answer: The regulatory actions related to station blackout (SBO) are Tier 1 activities that fall under mitigation strategies. For the latest information concerning mitigation strategies, which includes the mitigation strategies order and station-blackout mitigation-strategies rulemaking, please visit: http://www.nrc.gov/reactors/operating/ops-experience/japan-dashboard/mitigation-strategies.html.
Question No. 2
The USA presented a very detailed and comprehensive report. The identification of changes to the previous report is considered as a good practice and facilitates the peer review considerably.
Answer: Thanks for acknowledging this. We appreciate your comment.
Question No. 3
Degradation of buried piping systems: Can you elaborate the important aspects/considerations in the industry initiative to ensure the integrity of buried piping/control degradation in buried piping in USA?
Answer: The critical aspects of the buried piping industry initiative fall into three categories. First, the initiative is a phased program which includes the following concepts: • program development • risk ranking • inspection planning • conduct of initial inspections • development of an asset management plan This approach ensures that, irrespective of the status of a plant's buried piping program before the initiative, all plants will develop programs which are both similar in nature and effective. Second, the initiative has no end. The asset-management plans developed as part of the

initiative will address aging of buried pipe far into the future. Third, active engagement between the NRC and the nuclear industry in the implementation of the initiative has been highly beneficial. Documentation of NRC activities involving buried pipe is available on the NRC Web site at http://www.nrc.gov/reactors/operating/ops-experience/buried-piping-activities.html.
Question No. 4
Moisture effect on underground cables - As understood from the report, the licensees applying for a 20 year license renewal, have agreed to implement a cable testing program during the period of extended operation. However, only a few plants have established a cable testing program for the current operating period. Does NRC have plan to enforce cable testing programe in the plant for which license is already issued (40 year) and if so, what are the measures/key elements to be considered?
Answer: During the 40-year operational period, licensees are required to monitor the performance of equipment such as cables in accordance with Title 10, "Energy," of the *Code of Federal Regulations* (10 CFR) Section 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," (i.e., the Maintenance Rule) and program requirements in Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities." More specifically, Criterion XI, "Test Control," of Appendix B to 10 CFR 50 requires that a test program be (1) established to ensure that all testing required to demonstrate that structures, systems, and components (SSCs) will perform satisfactorily in service and (2) performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents. As also stated in the U.S. Sixth National Report, the NRC's Reactor Oversight Process (ROP) baseline inspection program routinely inspects maintenance and testing programs to identify any performance issues. To date, NRC inspectors have identified violations of requirements of Appendix B to 10 CFR Part 50 associated with performance monitoring of cables at several nuclear facilities. These violations are entered into the plant's corrective-action program and addressed accordingly.
Question No. 5
As mentioned in the report degradation in concrete was observed due to alkali-silica reaction at Seabrook station which was constructed in late 1970. It was primarily because the concrete mix has susceptible aggregate that was slow reacting. It was stated that potential reactivity of this aggregate was undetected by the testing specified by applicable ASTM construction standard at the time of construction. Were similar cases of degradations observed in other plants of USA of the same vintage or it is an isolated case.
Answer: No other cases of alkali-silica reaction degradation have been reported in the U.S. nuclear fleet. To the NRC staff's knowledge the degradation at Seabrook is an isolated case.
Question No. 6
Ireland would like to commend the USA on its comprehensive national report.
Answer: Thanks for acknowledging this. We appreciate your comment.
Question No. 7
The "Grow your own Probabilistic Risk Assessment Analyst Program" referred to on page 49 is an interesting concept. Has there been any resistance within Agency offices in releasing candidates from their current roles or making them available to complete the 3 year training program? How does NRC plan to retain these trained personnel once the program is complete?

Answer: The NRC recruited for the Grow Your Own Probabilistic Risk Assessment Analyst Program from internal NRC and external sources. Successful completion of the program results in a promotion for some employees up to a certain level. For NRC employees who seek a promotion, our internal policies, for the most part, require their release from the originating organization without resistance. Once a candidate is selected by the new organization, the NRC imposes no retention requirements for those who wish to leave before, or after, completion of the program. Given that fact, we realize that there will be an attrition rate of Grow Your Own candidates based on other job opportunities that might arise. Our hiring has reflected that. However, in light of future regulatory opportunities, we feel that the majority of candidates will have the incentive to advance in their careers through the probabilistic risk-assessment (PRA) path within the agency.

Question No. 8

The NRC issued an RFI [request for information] requiring each reactor to reevaluate the seismic and flooding hazards at its site using present-day methods and information, conduct walkdowns of its facilities to ensure protection against the hazards in its current design-basis, and assess its emergency communications systems and staffing levels. Licensees began submitting responses early in 2013 and the NRC staff is continuing to evaluate the responses.

Could you show us of the main summary of the evaluation? Especially could you show us of the good practices

Answer: For the latest regulatory related information on seismic reevaluations, please visit http://www.nrc.gov/reactors/operating/ops-experience/japan-dashboard/seismic-reevaluations.html.

For the latest regulatory related information on flooding reevaluations, please visit http://www.nrc.gov/reactors/operating/ops-experience/japan-dashboard/flooding.html.

For the latest regulatory related information on seismic and flooding walkdowns, please visit: http://www.nrc.gov/reactors/operating/ops-experience/japan-dashboard/walkdowns.html

For the latest information on emergency communications systems and staffing levels, please visit http://www.nrc.gov/reactors/operating/ops-experience/japan-dashboard/emerg-preparedness.html

INPO has issued the following documents to its members:

- INPO 11-005, Addendum, “Lessons Learned from the Nuclear Accident at the Fukushima Daiichi Nuclear Power Station,” August 2012 (publicly available at the Nuclear Energy Institute’s (NEI’s) Web site as http://www.nei.org/corporatesite/media/filefolder/INPO_11-005_Fukushima_Addendum_1.pdf).
- INPO 11-009, “Fukushima Forum Proceedings,” December 2011. (Document not public. Available to INPO members and participants only.)
- INPO 11-005, “Special Report on the Nuclear Accident at the Fukushima Daiichi Nuclear Power Station,” December 2012. (Document not public. Available to INPO members and participants only.) http://www.nei.org/corporatesite/media/filefolder/11_005_Special_Report_on_Fukushima_Daiichi_MASTER_11_08_11_1.pdf.
- INPO Event Report (IER) L1-11-1, Supplement 1, “Fukushima Daiichi Nuclear Station Fuel Damage Caused by Earthquake and Tsunami,” October 3, 2011. (Document not

public. Available to INPO members and participants only.)

- IER L1-11-4, "Near-Term Actions to Address the Effects of an Extended Loss of All AC Power in Response to the Fukushima Daiichi Event," revised September 29, 2011. (Document not public. Available to INPO members and participants only.)
- IER L1-11-2, "Fukushima Daiichi Nuclear Station Spent Fuel Pool Loss of Cooling and Makeup," April 25, 2011. (Document not public. Available to INPO members and participants only.)
- IER L1-11-1, "Fukushima Daiichi Nuclear Station Fuel Damage Caused by Earthquake and Tsunami," revised March 18, 2011. (Document not public. Available to INPO members and participants only.)

Question No. 9

Buried piping. A lot has been started by the operators. What would have been the regulatory action if there had not been voluntary action from the industry? And what enforcement options does NRC have, if there is a voluntary initiative?

Answer: It is difficult to speculate on precisely what actions would have been taken by the NRC had the industry not developed its initiative. It is possible, but not certain, that regulations would have been developed to accomplish the objectives which are currently being met by the industry initiative.

The NRC has no regulatory authority to directly enforce compliance with the initiative. The NRC does, however, retain normal programmatic enforcement authority related to issues such as following plant procedures. As such, the NRC would have enforcement authority if a plant had implementation procedures for the initiative but failed to follow them.

It should be noted that while the industry initiative is voluntary with respect to the NRC, it is mandatory within the U.S. nuclear industry. Approval from the industry, not just approval at the plant or corporate level, is required for a plant to deviate from the initiative. Since the inception of the initiative, only one such deviation has been taken. To date, the NRC has found that frequent meetings with industry are a valuable tool in ensuring the effectiveness of the implementation of the initiative.

Question No. 10

LTO [long-term operation] programme – Why was neutron absorber degradation not part of the LTO programme ? The same question regarding moisture effect on Underground cables.

Answer: Both neutron-absorber degradation and the moisture effect in underground cables are part of the overall aging-management strategy. Aging-management programs (AMPs) XI.M22, "Boraflex Monitoring," and XI.M40, "Monitoring of Neutron-Absorbing Materials Other Than Boraflex," in the "Generic Aging Lessons Learned (GALL) Report" (NUREG-1801, Revision 2, 2010) provide guidance on how to age-manage the degradation of neutron absorbers; the moisture effect for underground cables is addressed in AMP XI.E3 in the same document.

Question No. 11

Hardened vents, hydrogen control and mitigation and emergency preparedness enhancements for prolonged SBO and multi-unit events are identified by the international nuclear community as necessary improvements in the light of the Fukushima Dai-Ichi accident. For the last item in many countries measures are already implemented, at least as temporary measures, like installed mobile equipment.

Have these been implemented in the USA as well, noting they seem to be classified as 'Tier 3'

and thus may be delayed? We note that with 'Tier-1' some SBO-actions seem to have been listed.
Answer: The NRC agrees that hardened vents, hydrogen control, and emergency-preparedness enhancements for prolonged station blackout (SBO) and multi-unit events are important issues to be addressed as a result of the lessons learned from the Fukushima Dai-ichi accident. As noted in the question, actions to address these items are in the NRC Fukushima lessons-learned process. Tier 1 consists of actions that should begin without unnecessary delay. Tier 3 consists of actions that require further staff study to support regulatory action; need the result of an associated short-term action to inform the long-term action; depend on the availability of critical skill sets; or relate to potential revisions to the regulatory framework that balances defense-in-depth and risk considerations. For new reactor designs, the NRC is evaluating those designs with respect to the Tier 1 actions. These evaluations include assessing the ability to cope with prolonged SBO. The passive design features incorporated in some new reactors offer safety benefits with respect to coping for SBO conditions. New reactor designs will also be considered as the Tier 3 actions are studied. Requirements for reliable hardened vents for Mark I and Mark II containments are addressed as a Tier 1 activity; while requirements for reliable hardened vents for other containment designs are addressed as Tier 3. Emergency-preparedness enhancements for prolonged SBO and multi-unit events are addressed as Tier 3 activities. On March 12, 2012, the NRC issued Orders and 10 CFR 50.54(f) letters as directed by the Commission in staff requirements memorandum (SRM-)SECY-12-0025: • Enforcement Action (EA-)12-050: “Reliable Hardened Vents, BWR [Boiling-Water Reactor] Mark I and Mark II Containments” • EA-12-049: “Mitigation Strategies” On June 6, 2013, the NRC issued Revised Order EA-13-109 requiring severe-accident-capable hardened vents for BWR Mark I and Mark II containments. For more details on hardened vents, please visit http://www.nrc.gov/reactors/operating/ops-experience/japan-dashboard/hardened-vents.html Three aspects of Emergency Preparedness addressed for multi-reactor and loss-of-power events include: 1. training and exercises (drills)—consolidated into mitigating strategies 2. equipment, facilities, and related resources—consolidated into mitigating strategies 3. multiunit dose-assessment capability—NRC-endorsed industry initiative For more details on mitigating strategies, please visit http://www.nrc.gov/reactors/operating/ops-experience/japan-dashboard/mitigation-strategies.html. For more details on industry initiatives and NRC endorsement, please visit http://www.nrc.gov/reading-rm/doc-collections/commission/comm-secy/2013/2013-0010comscy.pdf and http://www.nrc.gov/reading-rm/doc-collections/commission/comm-secy/2013/2013-0010comsrm.pdf.

Question No. 12
Why is USA classifying ‘implementing Reactor Oversight Process modifications to reflect the recommended defense-in-depth framework’ (post-Fukushima) as Tier-3?

Answer: ROP modifications to reflect the recommended defense-in-depth framework is a Tier 3 recommendation because it depends on the resolution of Recommendation 1, which consists of an overall recommendation and four sub-recommendations. The overall recommendation is to establish a “logical, systematic, and coherent regulatory framework for adequate protection that appropriately balances defense-in-depth and risk considerations.” The four sub-recommendations are:

1. Draft a Commission policy statement that articulates a risk-informed defense-in-depth framework that includes extended design-basis requirements in the NRC’s regulations as essential elements for ensuring adequate protection.

2. Initiate rulemaking to implement a risk-informed defense-in-depth framework consistent with the above recommended Commission policy statement.

3. Modify the regulatory analysis guidelines to more effectively implement the defense-in-depth philosophy in balance with the current emphasis on risk-based guidelines.

4. Evaluate the insights from the individual plant examination and individual plant examination of external events efforts as summarized in NUREG-1560, “Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance,” issued December 1997, and NUREG-1742, “Perspectives Gained from the IPEEE [Individual Plant Examination for External Events] Program,” issued April 2002, to identify potential generic regulations or plant-specific regulatory requirements.

In SRM-SECY-11-0093 (August 19, 2011), the Commission set forth its direction to the staff with respect to the recommendations in the Near-Term Task Force (NTTF) report. For Recommendation 1, the Commission stated: “Recommendation 1 should be pursued independent of any activities associated with the review of the other Task Force recommendations. Therefore, the staff should provide the Commission with a separate notation vote paper within 18 months of the issuance of this SRM. This notation vote paper should provide options and a staff recommendation to disposition this Task Force recommendation.”

On December 6, 2013, the NRC staff presented to the Commission SECY-13-0132 with three potential regulatory improvement activities to disposition NTTF Recommendation 1. These potential improvement activities were developed after evaluation of the considerations underlying the NTTF’s recommendation and consideration of the NRC’s Risk-Management Task Force power-reactor recommendations. These activities constitute practical, low-cost improvements that can be implemented while consideration is given to other safety and regulatory initiatives such as the Risk-Management Regulatory Framework. The improvement activities, which are being considered by the Commission, are:

1. Establish a design-basis extension category of events and requirements and associated internal NRC guidance, policies, and procedures. The design-basis extension category would be applied in a forward-looking and generic basis. The internal NRC guidance would specify how to write future design-basis extension requirements in a consistent, logical, and complete manner, including the need to address “attributes” such as performance goals, treatment requirements, documentation requirements, change

processes, and reporting requirements.

2. Establish Commission expectations for defense in depth through the development of a policy statement that includes: the definition, objectives, and principles of defense in depth; associated implementation guidance containing decision criteria for ensuring adequacy of defense in depth; and conforming guidance to ensure integration of defense in depth with risk.

3. Clarify the role of voluntary industry initiatives in the NRC regulatory process by specifying when these initiatives may be credited and providing guidance regarding what type and level of licensee documentation and NRC oversight is appropriate for future industry initiatives.

Question No. 13

After the findings of flaws in the RPVs at the Doel and Tihange NPPs [nuclear power plants], European plants did additional ultrasonic testing. They do not seem to be required in the USA. What is your view on this?

Answer: The NRC staff has been working closely with the U.S. nuclear industry to evaluate the possibility of fabrication flaws of the type observed at Doel and Tihange existing in U.S. reactor pressure vessels (RPVs), as well as the potential safety implications of such flaws. The U.S. industry reviewed original fabrication records for the U.S. reactor pressure vessels and performed a probabilistic fracture-mechanics analysis to assess the likelihood of core damage assuming the presence of the type of quasi-laminar indications observed in the Doel 3 and Tihange 2 vessels. The industry presented its review during its Annual Industry/U.S. NRC Materials Programs Technical Information Exchange Meeting, which took place from June 5 through 7, 2013 (http://pbadupws.nrc.gov/docs/ML1316/ML13161A186.html), and presented to the NRC staff during a December 5, 2013, public meeting. Based on the preliminary industry results, the NRC did not find a need to mandate ultrasonic testing for this type of flaws at this time. The NRC will review the formal written report on the results of the industry's evaluations and will determine whether there is need for future regulatory action based on its formal review. The NRC has published Information Notice (IN) 2013-19 to inform U.S. licensees about this issue (http://pbadupws.nrc.gov/docs/ML1324/ML13242A263.pdf).

Question No. 14

NRC staff is assessing possible changes to regulations through the rulemaking process to include strategies for filtering or otherwise confining radioactive material that gets released as a reactor core is damaged, i.e. filtered venting.
How many of the NPP's in the US are already equipped with filtered containment venting systems?

Answer: The U.S. nuclear power plants are not currently equipped with a hardened filtered containment venting system. BWR plants with Mark I containments installed hardened containment vent systems in response to Generic Letter (GL) 89-16. As a result of Fukushima, all Mark I and Mark II containment design nuclear power plants are now required to have a reliable hardened containment vent that is capable of functioning under severe accident conditions. The need for the capability to filter the releases by an external filter or other strategies is currently under evaluation (also see answer to questions no. 156 and 179). Most U.S. plants currently have the ability to line up and vent containment of noncombustible and low airborne contaminated gases through a filtered low-pressure low-flow-rate standby gas treatment system or other filter trains during normal plant operations.

Question No. 15
USA has invited a few OSART missions in three years time, having a fleet of over 100 reactors. In the post-Fukushima IAEA-Action Plan it is recommended that Each Member State with nuclear power plants will voluntarily host at least one IAEA Operational Safety Review Team (OSART) mission during the coming three years, with the initial focus on older nuclear power plants. Thereafter, OSART missions are to be voluntarily hosted on a regular basis. If on a regular basis means every 10 years and this would be the practice in USA, this would mean that USA has to invite 10 OSART-missions every year. What is your view on the necessity of such a periodicity?
<u>Answer</u>: The U.S. has long supported the OSART program and continues to do so. The NRC stresses the importance of peer reviews to its licensees and encourages their participation in the OSART program though NRC cannot compel licensee participation. Additionally, the U.S. nuclear industry has a review program through INPO, which is a non-governmental organization formed by the U.S. nuclear industry following the Three Mile Island accident. INPO conducts plant evaluations, provides industry training, performs event analyses and information exchanges, and provides assistance to nuclear plants on an as-requested basis. While INPO's activities do not replace the NRC's oversight function, INPO does provide the industry with a means of self-assessment that complements the NRC's role. INPO performs onsite plant evaluations approximately once every two years for every U.S. operating nuclear power plant. INPO evaluations are based on U.S. requirements and practices. It should also be noted that every six years each INPO evaluation is conducted as a World Association of Nuclear Operators peer review and includes at least four international peers. OSART missions are based on International Atomic Energy Agency (IAEA) Standards, which are similar to NRC regulations. The U.S. has been hosting an OSART mission every 3 years and participating in OSART missions in other countries annually. The U.S. believes that this level of onsite engagement is appropriate as part of assuring harmonization with international operational standards.
Question No. 16
Who many in total key safe operation indicators are used for assessment of safe operation of power units in addition to the WANO's indicators; what areas do they cover? How quality of maintenance and repair is assessed (are there indicators to run such an assessment)? Are there indicators, which assess leak-tightness of the reactor coolant circuit; if yes, what are they (methodology of their calculation)?
<u>Answer</u>: The NRC uses a total of 17 performance indicators that provide a broad sample of data to partially assess licensee performance in the risk-significant areas of each cornerstone of safety. These performance indicators are not directly aligned with the World Association of Nuclear Operators' (WANO's) performance indicators and are not intended to provide complete coverage of every aspect of plant design and operation. The NRC's performance indicators program is described in Inspection Manual Chapter (IMC) 0608, "Performance Indicator Program." Within three strategic performance areas, performance indicators were developed for each of the safety cornerstones to provide an objective indication of licensee performance. The strategic performance areas, cornerstones of safety, and their associated performance indicators are as follows: Strategic Performance Area: Reactor Safety 1. Initialing Events Cornerstone: The objective of this cornerstone is to limit the frequency

of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The performance indicators for this cornerstone are as follows:

- Unplanned Scrams per 7,000 Critical Hours—Monitors the number of unplanned scrams (manual and automatic). It measures the rate of scrams per year of operation and provides an indication of initiating event frequency.
- Unplanned Scrams with Complications—Monitors that subset of unplanned automatic and manual scrams that require additional operator actions and are more risk-significant than uncomplicated scrams.
- Unplanned Power Changes per 7,000 Critical Hours—Monitors the number of unplanned power changes (excluding scrams) that could have, under other plant conditions, challenged safety functions.

2. Mitigating Systems Cornerstone: The objective of this cornerstone is to ensure the availability, reliability, and capability of systems that mitigate plant transients and the reactor accidents. The performance indicators for this cornerstone are as follows:

   - Mitigating System Performance Index—Monitors the readiness of important safety systems to perform their safety functions in response to off-normal events or accidents. The systems include high-pressure injection, heat removal, residual heat removal, emergency alternating-current power, and support cooling water. Each of the above systems is represented by a single performance indicator.
   - Safety System Functional Failure—Monitors events or conditions that could have prevented the fulfillment of the safety function of structures or systems that are needed to (1) shut down the reactor and keep it in a safe condition, (2) remove residual heat, (3) control the release of radioactive material, and (4) mitigate the consequences of an accident.

3. Barrier Integrity Cornerstone: The objective of this cornerstone is to ensure that physical barriers protect the public from radionuclide releases caused by accidents. The performance indicators for this cornerstone are as follows:

   - Reactor Coolant System Specific Activity—Monitors the integrity of the fuel cladding.
   - Reactor Coolant System Leakage—Monitors the integrity of the reactor coolant system pressure boundary.

Strategic Performance Area: Radiation Safety

1. Emergency-Preparedness Cornerstone: The objective of this cornerstone is to ensure that actions taken by the emergency plan would provide protection of the public health and safety during a radiological emergency. The performance indicators for this cornerstone are as follows:

   - Alert and Notification System Reliability—Monitors the reliability of the offsite Alert and Notification System, a critical link for alerting and notifying the public of the need to take protective actions.
   - Emergency Response Organization Drill Participation—Measures the percentage of key Emergency Response Organization members who have participated recently in drills and exercises or in an actual event.
   - Drill/Exercise Performance—Monitors timely and accurate licensee performance in

drills and exercises when licensees/operators are presented with opportunities for classification of emergencies, notification of offsite authorities, and development of protective-action recommendations.

2. Public Radiation Exposure Cornerstone: The objective of this cornerstone is to ensure adequate protection of public health and safety from exposure to radioactive material released into the public domain as a result of routine civilian nuclear-reactor operations. The performance indicator for this cornerstone is as follows:

   - Radiological Technical Specifications/Offsite Dose Calculation Manual Radiological Occurrence—Assesses the performance of the radiological effluent-control program.

3. Occupational Radiation Safety Cornerstone: The objective of this cornerstone is to ensure adequate protection of worker health and safety from exposure to radiation from radioactive material during routine civilian nuclear-reactor operation. The performance indicator for this cornerstone is as follows:

   - Occupational Exposure Control Effectiveness—Monitors the control of access to and work activities within radiologically significant areas of the plant and occurrences involving degradation or failure of radiation safety barriers that result in readily identifiable unintended dose.

Strategic Performance Area: Safeguards

1. Security Cornerstone: The objective of the security cornerstone is to provide assurance that the licensee's security system and material control and accounting program use a defense-in-depth approach and can protect against (1) the design-basis threat of radiological sabotage from external and internal threats and (2) the theft or loss of radiological materials. The performance indicator for this cornerstone is as follows:

   - [Performance indicators pertaining to the Security Cornerstone will not be publicly available to ensure that security information is not provided to a possible adversary.]

Performance indicators within the Reactor Safety cornerstones assess quality of maintenance and repair at a facility. The failure of a risk-significant valve or pump as a result of poor quality of maintenance and repair would be reported within the Mitigating Systems Performance Index or the Safety System Functional Failure performance indicator. Additionally, if the equipment failure resulted in a scram or significant reactor power decrease, the event would be reported within the Unplanned Scrams and/or Unplanned Power Changes per 7,000 Critical Hours performance indicator. Performance-indicator values are reported by the licensee for each reactor unit and assessed by the NRC on a three-month frequency (that is, quarterly). These values are then compared to performance thresholds established for each performance indicator. Licensee performance relative to these thresholds is assessed together with inspection findings to inform the NRC's regulatory response.

The Barrier Integrity cornerstone contains two performance indicators that monitor two of the three physical design barriers to prevent the release of radioactive fission products. Performance indicators for the reactor coolant system activity and reactor coolant system leakage monitor the integrity of the fuel cladding barrier and reactor coolant system respectively. The reactor coolant system's leakage indicator specifically assesses the leak-tightness of the reactor coolant system. The calculation uses the maximum reactor coolant identified leakage each month divided by the plant's technical specification limit. The final value is expressed as a percentage of the leakage allowed by the plant's technical specifications. The limits that inform the NRC response are set at 50 percent and 100 percent of the plant's specification limit.

Question No. 17

How are corrective measures and good practices implemented (at NPPs and corporate level), basing on results of the analysis of safe operation indicators?

Answer: NRC regulations require all licensees to develop and maintain a corrective-action program. The corrective-action program is the system by which a utility finds and fixes problems at the nuclear plant. It includes a process for evaluating the safety significance of the problems, setting priorities in correcting the problems, and tracking them until they have been corrected. A basic tenet of the current ROP is that a licensee's corrective-action program should be (1) relied on to correct identified issues, and (2) effective to prevent reoccurrence. The founders of the oversight process theorized that the breakdown of a corrective-action program would eventually manifest itself through a performance decline that would be identified through cornerstone performance indicators and/or inspection findings. However, during development of the current ROP, some individuals expressed concern that corrective-action program effectiveness might not be adequately evaluated using performance indicators and the inspection program. There was a concern that significant issues might exist even when performance indicator and inspection finding thresholds are not crossed. Therefore, focused inspections of licensee corrective-action programs are conducted routinely under the baseline inspection program.

Question No. 18

According to the IAEA's Power Reactor Information System (PRIS), in the USA there are reactors with different fuel cycles (18, 24) in operation. The National Report (Appendix 2) provides an assessment of the indicators by years.
Is there experience in analyzing the indicators used for assessment of safe operation basing on the fuel cycle duration? If yes, please, describe.

Answer: In the United States, we find it useful to use cycle values to compare performance between units for indicators that are significantly influenced by refueling outages, such as unit capability factor and collective radiation exposure. However, our analyses have not indicated that one fuel-cycle duration results in safer operation than the other.

Question No. 19

Appendix 2 provides values of the Unit Capability Factor for the period 2003-2012.
What is the reason for such high and stable values? What is an average duration of overhauls at NPPs with different reactors?

Answer: [Regarding Annex 2] Consistently higher unit capability factor values are the result of lower forced-loss rates. The occurrence of fewer unplanned outages, as a result of focusing on critical equipment performance during online and outage periods, has proved more of a factor in improving unit capability factor than better outage performance. Other factors include:

- At U.S. utilities, we conduct online maintenance.
- We focus on single-point vulnerability.
- We perform preventive maintenance optimization.
- We focus on eliminating plant trips.

Typical U.S. refueling outages are completed in 30 to 35 days.

Question No. 20
Do you have any experience with degradation or deformation of steel type of Boral?
Answer: The NRC is not aware of any documented degradation or deformation of the borated stainless steel neutron absorbing material in the spent fuel pool. However, because future degradation might occur, many sites have implemented monitoring programs.
Question No. 21
What is obligation for licensees applying for a 20-year license renewal, regarding RG 1.218?
Answer: Licensees have no obligation to implement cable condition-monitoring techniques outlined in Regulatory Guide (RG) 1.218, “Condition-Monitoring Techniques for Electrical Cables Used in Nuclear Power Plants.” RG 1.218 provides an acceptable means to implement condition monitoring to meet the requirements of 10 CFR Part 50 for periodic inspection and testing to monitor the performance of electrical cables. Licensees may propose alternative techniques and conditional monitoring programs if the alternative techniques or programs are deemed acceptable to the NRC.
Question No. 22
What is the most challenging activity to discover CFSI? Which industry guidelines for effectively quarantining suspect parts already exist?
Answer: The biggest challenge in discovering counterfeit, fraudulent, and suspect items (CFSIs) is the workmanship used to avoid detection. Receipt inspection skills and detection techniques improve as organizations become more aware of CFSI threats. However, the skills and craftsmanship employed to avoid detection are also improving. Modern technological advances in computer-aided manufacturing and graphic arts increase the difficulty of detecting CFSI. These advances are most prevalent in the counterfeiting of electronic components, where sophisticated testing and inspection techniques are essential for detecting counterfeit components. Guidance for retaining CFSI can be found in the U.S. Department of Energy (DOE) Publication DOE G 414.1-3, Section 6.2.7, “How to Secure S/CI,” and in Electric Power Research Institute (EPRI) Document No. 1019163, “Plant Support Engineering: Counterfeit, Fraudulent, and Substandard Items – Mitigating the Increasing Risk,” Section 7.9.3, “Disposition of Suspected or Confirmed CFSIs.” Guidance for the control of CFSI within the international commercial nuclear industry can be found in IAEA TECDOC 1169, “Suspect Counterfeit Items,” Section 3.5.3, “Disposal of S/CIs.”
Question No. 23
1.3.2 Current Safety and Regulatory Issues - Regarding mitigation of cumulative effects of regulation, please give more details on the available tools to allow NRC focus on ítems of greatest safety impact.
Answer: The NRC currently uses the “Common Prioritization of Rulemaking” to prioritize (for budget purposes only) prospective and current rules by considering four factors and assigns a score to each factor. Factors include: support for strategic goals, support for excellence objectives, interest to the government, and interest to the members of public, non-governmental organizations, the nuclear industry, etc. The NRC is also exploring a prioritization initiative that could use risk information to determine priority of a regulatory action, both generically and on a plant-specific level. This prioritization initiative is in its early planning stages. Lastly, the NRC obtains feedback on prospective rulemakings during the proposed rule’s public comment period and also during a public meeting on implementation that is held during the final rule stage. Specifically, the NRC asks the public to describe any ongoing regulatory actions that could impact the implementation of the prospective rulemaking. This information is then used to determine the optimal implementation schedule for the prospective rulemaking.

Question No. 24
One of the principal insights of the Fukushima accident is the need for a reliable ultimate heat sink (UHS). The Swiss regulator, as well as ENSREG [European Nuclear Safety Regulators Group], considered this fact in their decisions to issue recommendations (ENSREG) and orders (ENSI [Swiss Federal Nuclear Safety Inspectorate]) to implement a diversified UHS for NPPs which are not already provided with this important safety feature. In several western european countries, the corresponding ENSREG recommendation has already been implemented. Could you please give information on the NRCs and the industry's view of this issue? Are there any considerations to implement a diversified heat sink in the short- or medium term?
Answer: The NRC's SBO Mitigation Strategies rulemaking effort will codify into the agency's rules the requirements already imposed by the Mitigation Strategies Order that was issued by the NRC on March 12, 2012. The rule will ensure that if a plant loses power, it will have sufficient procedures, strategies, and equipment to cope with the loss of power for an indefinite amount of time. Subsumed in the rulemaking is the need to maintain a reliable ultimate heat sink. The mitigation strategies are expected to use a combination of currently installed equipment (e.g., steam-powered pumps), additional portable equipment that is stored onsite, and equipment that can be flown in or trucked in from support centers. For more information on mitigation strategies, please visit http://www.nrc.gov/reactors/operating/ops-experience/japan-dashboard/mitigation-strategies.html.
Question No. 25
We acknowledge this table, as it provides a helpful overview for the reviewers and saves time. This is an example of good practice in reporting.
Answer: Thanks for acknowledging this. We appreciate your comment.
Question No. 26
"Degradation of Neutron-Absorber Materials in Spent Fuel Pools - One of the NRCs strategic outcomes for its safety goal is that there are "no inadvertent criticality events." Since the licensee did not have an established monitoring program for the Carborundum, the onset of the degradation and the degradation rate cannot be established. The safety significance of this finding is that the licensees SFP was in an unanalyzed condition and, although margin remained, the amount of margin to the pool having a criticality event was not known. " Has the monitoring of criticality been influenced by events in Fukushima? Are cases considered where for instance, the rods are starting to become uncovered and a water mist is formed by the evaporation of the remaining water, would this lead to a cliff-edge effect in terms of criticality?
Answer: The monitoring of criticality and the types of criticality events that licensees consider in the spent fuel pool have not been influenced by the events in Fukushima. The types of criticality events that a licensee considers in the spent fuel pools have not changed and include misloading, assembly drops, and boron-dilution scenarios.
Question No. 27
There is no mention of the need to diversify the heat sink in the Tier 1/2/3 recommendations, nor the danger of internal flooding. Has any thought been given to these matters?
Answer: Diversified heat sink capability is being addressed within the Tier 1 mitigation-strategies effort, in which licensees are expected to cope with an extended loss of electrical power (from a variety of initiating events) for an indefinite amount of time using a combination of currently

installed equipment (e.g., steam-powered pumps), additional portable equipment that is stored onsite, and equipment that can be flown in or trucked in from support centers. With regards to internal flooding, the Tier 3 activity related to seismically induced fires and floods will consider the potential effects of internal flooding from seismically induced failure of pipes, tanks, etc. Although, in the early 1990s, U.S. licensees completed individual plant examinations (which were probabilistic analyses that estimated core-damage frequency and containment performance for accidents initiated by internal events, including internal flooding), the Commission did not require changes as a result of the individual plant examination program, but strongly encouraged plants to make improvements based on results. NRC publication NUREG-1560 documents the individual plant examination program and its conclusions. The current Tier 1 effort on mitigation strategies is expected to provide significant capabilities that can be used to respond to internal flooding events as well.

For more information on mitigation strategies, please visit http://www.nrc.gov/reactors/operating/ops-experience/japan-dashboard/mitigation-strategies.html.

Question No. 28

The Atomic Energy Act and NRC regulations limit commercial power reactor licenses to 40 years but permit such licenses to be renewed. The Atomic Energy Act established the original 40-year term, a timeframe based on economic and antitrust considerations, rather than the technical limitations of the nuclear facility.

The 40 year timeframe has had a major impact on regulators worldwide. Generally, outside the USA, the economic rational for this decision is not known. Is it to be understood from this extract that there is no 40 year technical limit built in to the design of US power plants from the point of view of the regulator?

Answer: The 40-year maximum timeframe is established in section 103(c) of the Atomic Energy Act of 1954, as amended (AEA, Public Law 83-703, http://pbadupws.nrc.gov/docs/ML1327/ML13274A489.pdf), which states that each license "shall be issued for a specified period…not exceeding forty years from the authorization to commence operations and may be renewed on the expiration of such period." The selection of the 40-year maximum timeframe was based on economic and antitrust considerations, not technical considerations. The NRC has determined that nuclear reactors can be operated safely beyond 40 years and has established a license renewal program allowing 20-year extensions following a technical review of the applicants' programs for managing the adverse effects of aging.

Question No. 183

In the summary it was mentioned that there is a Plant Parameter Envelope (PPE) process. Please provide further information on how the PPE relates to the licensing process under 10CFR52 respectively 10CFR50?

Answer: As part of the licensing process under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," the NRC can issue an early site permit to approve a site for a nuclear power plant independent of an application for a construction permit (Part 50) or a combined license (Part 52), and independent of a selected certified design. 10 CFR Part 52 does not require an applicant to provide specific design information. However, some design information might be required to address 10 CFR 52.17(a)(1), which calls for "an analysis and evaluation of the major SSCs of the facility that bear significantly on the acceptability of the site under radiological consequence evaluation factors identified in 10 CFR 52(a)(1)(ix)(A) and 10 CFR 52(a)(1)(ix)(B)."

The plant-parameter envelope approach has been chosen by most early site permit applicants as a means for ensuring that necessary design information is provided in an application to allow the NRC to make a licensing decision. The plant-parameter envelope is a set of postulated design parameters listed in an application that are expected to bound the characteristics of multiple reactor designs that might later be selected for a site. While an early site permit is intended to provide finality on site safety issues, environmental-protection issues, and plans for coping with emergencies, a construction-permit or combined-license applicant citing an early site permit must address any licensing issues that were not resolved as part of the cited early site permit proceeding before the NRC can issue a construction permit or a combined license.
Question No. 184
The summary describes the activities related to NRCs oversight and evaluation of the degradation of neutron absorbing materials in spent fuel pools. The report identifies the observed degradation as well as uncertainties in some surveillance methodologies used. Please provide references for the Technical Letter Reports related to uncertainties in surveillance methodologies mentioned on page 23. The National Report mentions that additional reports pertaining to other aspects of neutron absorbing-material degradation will be issued in the near future. Please identify the topics that will be covered in these future reports.
Answer: The Technical Letter Reports cited on page 23 are: • "Boraflex, RACKLIFE, and BADGER [Boron Areal Density Gauge for Evaluating Racks]: Description and Uncertainties" (ADAMS Accession No. ML12216A307) • "Initial Assessment of Uncertainties Associated with BADGER Methodology" (ML12254A064) • "Monitoring Degradation of Phenolic Resin-Based Neutron Absorbers in Spent Nuclear Fuel Pools" (ML13141A182) These documents are available on the NRC's public Web site at http://www.nrc.gov/waste/spent-fuel-storage/pools.html. Other topics that might be covered in future technical letter reports might include additional reports on other neutron-absorbing materials such as Boral and other metal matrix composites.
Question No. 185
The summary describes the issue of containment pressure credit for NPSH in some reactor safety analyses and notes that the Commission directed staff to assure that the defense-in-depth philosophy is interpreted and implemented consistently in RG 1.174. Please elaborate on the draft guidance and the particular risks related to operation of plants of this design.
Answer: As part of reactor safety calculations, licensees must demonstrate that the Emergency Core-Cooling System (ECCS) pumps and containment heat-removal pumps will perform their safety functions of (1) delivering the cooling flow, as required by General Design Criterion (GDC) 35, "Emergency Core Cooling," in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 and (2) rapidly reducing the containment pressure and temperature, as required by GDC 38, "Containment Heat Removal." The ECCS pumps must perform their safety function during a loss-of-coolant accident in order to satisfy the requirements of 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors." NRC RG 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps," dated November 2, 1970, states that the pressure in containment before the postulated accident should be used when determining the available net positive

suction head (NPSH) of ECCS and containment heat-removal system pumps. Position 1.3.1.1 in Revision 4 of RG 1.82, “Water Sources for Long-Term Recirculation Cooling Following a Loss-Of-Coolant Accident”, states that the ECCS and containment heat-removal system should be designed so that adequate available NPSH is provided to the system pumps with no increase in containment pressure from that present before the postulated loss-of-coolant accidents. However, Position 1.3.1.2 in Revision 4 of RG 1.82 states that for operating reactors “for which the design cannot be practicably altered,” it is acceptable to use containment accident pressure greater than the containment pressure before the accident in determining the available NPSH of ECCS and containment heat-removal pumps.

For the purpose of demonstrating that the pumps will perform their safety function, licensees are required to demonstrate that the ECCS and containment heat-removal pumps have adequate NPSH margin following the occurrence of a postulated accident. The calculation of NPSH margin is currently performed in accordance with the positions of Revision 3 of RG 1.82, which specify conservative input values and worst-case assumptions, including the postulated accident scenario, intended to minimize the NPSH margin.

The NRC staff developed draft deterministic guidance (ADAMS Accession No. ML13015A437) to supersede portions of the guidance of Revision 3 of RG 1.82 after appropriate regulatory processes have been completed. The draft guidance considered and included Advisory Committee on Reactor Safeguards (ACRS) recommendations for quantifying uncertainty in the NPSH calculations.

In NRC Commission paper SECY-11-0014 (ML102590196), the staff presented options to the Commission to resolve outstanding issues related to the use of crediting containment accident pressure. The Commission provided direction to the staff in SRM-SECY-11-0014 (ML110740254). The Commission directed the staff to conduct the NPSH analysis in a way that credits containment accident pressure reviews consistently with current staff practice but also implements the new staff deterministic guidance. The Commission also directed the staff to assure that the defense-in-depth philosophy is interpreted and implemented consistently with RG 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis.” The draft guidance was sent to the Boiling Water Reactor Owners Group (ML13017A463) and the Pressurized Water Reactor Owners Group (ML13017A428) for comments and for proposing an implementation method if containment accident pressure is credited for calculating NPSH margin. The NRC staff has discussed the proposed draft guidance with both owners’ groups. The draft guidance will be issued as an Interim Staff Guidance (ISG) in the future.

The NRC staff has reviewed the objectives in Section 19.2 of NUREG-0800, “Standard Review Plan,” to ensure that a proposed license amendment maintains appropriate safety within the defense-in-depth philosophy. It is the staff’s position that crediting containment accident pressure in determining available NPSH is acceptable with respect to defense in depth. The NRC staff considers a risk-informed analysis which meets the guidance of RG 1.174 and Standard Review Plan Section 19.2 would successfully justify crediting containment accident pressure for determining the available NPSH of the ECCS and containment heat-removal pumps.

Question No. 186

The summary describes observations of steam generator tube wear at 7 or more NPPs with replacement steam generators between 2009-2012. Multiple SG tubes had indications of wear. Two of these plants have been permanently shut down. The section concludes that “no additional regulatory action has been deemed necessary at this time”. Please elaborate on how

have these observations affected NRC consideration of the consequential multiple SGTR [steam generator tube rupture] that was the subject of GSI-163, for example. Is this information used in the accident precursor program described elsewhere in the report? What are the impacts of the recent enhancements to SG tube surveillance programs (NEI 97-006) adopted by the US industry?
Answer: With the exception of eight indications of wear detected at San Onofre Nuclear Generating Station, Unit 3, all other wear detected in replacement steam generator (SG) tubes has met regulatory acceptance criteria. As a result, although a large number of indications of wear might be present at a few plants, it has not compromised the integrity of the reactor coolant pressure boundary. Therefore, the conclusions made in resolving Generic Safety Issue-163, “Multiple Steam Generator Tube Leakage,” are still valid and past risk studies on steam generators are still appropriate because no significant adverse generic trends have been identified. Because plants are required to ensure that the steam generator tubes have adequate integrity, if they were to find that they did not maintain integrity (as was the case for San Onofre, Unit 3), corrective action is required to be taken. Historically, actions taken in response to discovering a loss of tube integrity have been effective at ensuring that tube integrity is maintained in the future (for the condition that caused the loss of tube integrity). The industry initiative (NEI 97-06, “Steam Generator Program Guidelines,” dated December 1997) along with the replacement of steam generators has improved steam generator performance as evidenced by a decrease in forced outages caused by primary-to-secondary leakage and a reduction in the frequency of tube ruptures.
Question No. 187
The summary refers to the concrete degradation at Seabrook Station resulting from an alkali-silica reaction. Could the USA comment on the potential for generic occurrence of this phenomenon at other plants in the USA and elsewhere?
Answer: The susceptible slow-reacting aggregates that caused the alkali-silica reaction degradation at Seabrook are only common in certain regions of the United States (the Northeast and isolated areas of the Southwest). It is possible that other nuclear plants in these regions could experience similar “delayed” alkali-silica reaction; however, to date, no other cases of alkali-silica reaction degradation have been reported in the U.S. nuclear fleet. To address the generic implications of the Seabrook operating experience, the NRC issued IN 2011-20, “Concrete Degradation by Alkali-Silica Reaction” (http://pbadupws.nrc.gov/docs/ML1122/ML112241029.pdf). This IN informs U.S. nuclear power plant licensees that the ASTM International standards used to identify susceptible aggregate during the construction of many U.S. nuclear plants might not accurately identify slow-reacting aggregates. The IN further explains what licensees can look for if alkali-silica reaction is suspected and what steps can be taken if alkali-silica reaction is identified.
Question No. 215
The report states “The Commission has seen sustained, strong interest in license renewal, which allows plants to operate up to 20 years beyond their current operating licenses”. Will the period of licence renewal (20 years) be applied to all existing plants or will it depend on the existing plants’ condition?
Answer: The NRC regulations allow the renewal of nuclear power plant licenses for up to an additional 20 years beyond the initial licensing period of 40 years. So far, all 73 renewed licenses were granted 20 additional years of operation. The licensee may voluntarily cease operation, as in the case of Kewaunee Power Station (see Section 6.2 of the U.S. National Report), before the end of its license period because of other factors. It is also conceivable that

a plant might need to terminate its operation before its license period because of its inability to meet aspects of regulations (e.g., 10 CFR 50.61, “Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events”).

Question No. 216

The U.S. national report provides much useful information on its response to the events following Fukushima. With regards to the updates to the crisis communication plan/capability of the U.S. NRC, could you provide examples of improvements and lessons learned?

Answer: The NRC’s Office of Public Affairs (OPA) made a number of additions and other changes to its crisis communication strategy based on lessons learned during the U.S. response to the nuclear accident at Fukushima. Some of the changes and additions include:

1. Development and management of a Public Inquiry Desk/Internal Call Center team to answer public calls during a crisis was tasked to the Office of Nuclear Security and Incident Response (NSIR) in recognition of the resource limitations of the Office of Public Affairs staff to respond to all media and all public calls during a real incident. OPA and NSIR are working with the Office of the Executive Director for Operations on a paper to developing and implementing a roster and process/procedure for such a team.

2. OPA increased the number of technical briefers on its roster in recognition of the workload posed by the Fukushima accident.

3. In recognition of the workload challenges of the Fukushima accident, OPA increased its roster of additional agency staff members who could join the OPA to provide support in a real event, and created a matrix that better outlines each staffer’s skill set.

4. Staffing protocol in the crisis communication strategy was changed to limit the use of regional public affairs officers to augment the headquarters' staff in a real event. During Fukushima, regional personnel were busy with “can this happen at my plant” calls and emails, and it is now recognized that their ability to support headquarters activities would be limited.

5. The use of social media to disseminate information and to monitor information being disseminated by others was expanded in the crisis communication strategy after Fukushima. An informal agreement was subsequently made with the U.S. Federal Emergency Management Agency (FEMA) to monitor social media for the NRC in a real event.

6. OPA made updates to the dark “Emergency Event Web page,” a web page that would be activated in a real, domestic event. It also initiated a “Rumors” page that would be activated in a real event to fight inevitable false rumors being spread primarily through social media (as occurred during the Fukushima incident).

Question No. 228

It is understood that the National Report of the USA for the 6th Review Meeting was submitted later than the submission due date. I wonder if there was any reason you could explain for that delay. As several leading Contracting Parties did not meet the due date and also the effectiveness of CNS has been on debates after Fukushima, it is concerned that these delay might indicate the decreasing respect for the spirit and obligation of CNS.

Answer: For the United States, the delay reflects a heightened respect for the spirit and obligation of the CNS and ensuring the rigor of our National Report in light of the lessons of the Fukushima Dai-ichi accident and significant actions being taken in response. The U.S. National Report undergoes a very rigorous development, review, and approval process. The regulatory portions of the report are developed primarily by the NRC. The operational aspects of the report are prepared by INPO. The draft report is circulated within the U.S. Government to several other agencies for review, such as the U.S. Department of State and the DOE.
The final draft of the report must be reviewed and approved in a vote by the NRC's Commissioners.

During this review and approval process, areas where the report could be strengthened were identified by the reviewers. The delay in submitting the U.S. National Report resulted from the staff's review and incorporation of those comments on the draft report.

ARTICLE 6. EXISTING NUCLEAR INSTALLATIONS

Each Contracting Party shall take the appropriate steps to ensure that the safety of nuclear installations existing at the time the Convention enters into force for that Contracting Party is reviewed as soon as possible. When necessary in the context of this Convention, the Contracting Party shall ensure that all reasonably practicable improvements are made as a matter of urgency to upgrade the safety of the nuclear installation. If such upgrading cannot be achieved, plans should be implemented to shut down the nuclear installation as soon as practically possible. The timing of the shutdown may take into account the whole energy context and possible alternatives, as well as the social, environmental, and economic impact.

This section explains how the United States ensures the safety of nuclear installations in accordance with the obligations in Article 6. It covers the reactor licensing and major oversight processes in the United States. This section also discusses programs for rulemaking, fire-protection regulation, decommissioning, research, programs for public participation, and lessons learned from Fukushima. The NRC posts the major results of assessments on the agency's public Web site at http://www.nrc.gov.

Question No. 29
It is mentioned that NRC concluded that spent fuel can be stored safely in SFPs [spent fuel pools] or in onsite independent spent fuel storage installations without significant environmental impacts for at least 60-years beyond the plant's licensed life (Article 6; Section 6.3.9; page 70). Could you please explain which were the criteria used to conclude this statement?
Answer: The conclusion about the environmental impacts of spent fuel storage, and associated rulemaking, are currently under revision. This revision is now being supported by an environmental impact statement (NUREG-2157; http://pbadupws.nrc.gov/docs/ML1322/ML13224A106.pdf) and proposed rule (http://pbadupws.nrc.gov/docs/ML1325/ML13256A004.pdf), which are expected to be completed in October 2014.
Question No. 30
Were there safety analysis procedures for US NPP as analog to European stress-tests?
Answer: The United States did not perform a stress test. The United States stayed within the regulatory processes and performed the following in response to the nuclear accident at Fukushima. The following major activities have been completed. March 11, 2011—The NRC staffed its Headquarters Operations Center on a 24/7 basis, first monitoring tsunami effects on the U.S. West Coast, and then supporting both domestic and international responses to the Fukushima Dai-ichi reactor accidents until May 16, 2011. The first of many NRC reactor experts were sent immediately to Japan as part of a U.S. Agency for International Development (USAID) mission. Others followed in the weeks and months after. March 18, 2011—NRC issued IN 2011-05, "Tohoku-Taiheiyou-Oki Earthquake Effects on Japanese Nuclear Power Plants," to inform operators of nuclear power plants of the effects of the Tohoku-Taiheiyou-Oki Earthquake on nuclear power plants in Japan (http://pbadupws.nrc.gov/docs/ML1108/ML110830824.pdf).

March 23, 2011—The Commission directed the staff to establish a senior-level agency task force to conduct a methodical and systematic review of our processes and regulations to determine whether the agency should make additional improvements to our regulatory system and make recommendations to the Commission for its policy direction.

March 23, 2011—Using Temporary Instruction 2515/183, "Followup to the Fukushima Daiichi Nuclear Station Fuel Damage Event," NRC resident inspectors began re-examining post-9/11 emergency equipment and related items at U.S. nuclear power plants, in light of details from the Fukushima accident (http://pbadupws.nrc.gov/docs/ML1107/ML11077A007.pdf).

March 31, 2011—The NRC issued IN 2011-08, "Tohoku-Taiheiyou-Oki Earthquake Effects On Japanese Nuclear Power Plants—For Fuel Cycle Facilities," to share details of the Fukushima accident with U.S. nuclear fuel facilities, and reminded those facilities of the need to properly analyze the effects of severe natural events (http://pbadupws.nrc.gov/docs/ML1108/ML110830824.pdf).

April 1, 2011—The NRC created the NTTF to examine lessons learned from the Fukushima accident.

April 29, 2011—Using Temporary Instruction 2515/184, "Availability and Readiness Inspection of Severe Accident Management Guidelines (SAMGs)," NRC resident inspectors began examining severe-accident management procedures and training at U.S. nuclear power plants (http://pbadupws.nrc.gov/docs/ML1107/ML11077A007.pdf).

May 11, 2011—NRC issued Bulletin 2011-01, "Mitigating Strategies," requiring nuclear power plants to provide information on post-9/11 emergency equipment, as well as how the plants ensure that strategies to use the equipment remain effective over time (http://pbadupws.nrc.gov/docs/ML1112/ML111250360.pdf).

July 12, 2011—The NRC's NTTF issued its report on lessons learned from Fukushima, SECY-11-0093, "Recommendations for Enhancing Reactor Safety in the 21st Century." The NTTF concluded that U.S. plants were operating safely and provided 12 broad recommendations to the Commission for enhancing reactor safety (http://pbadupws.nrc.gov/docs/ML1118/ML111861807.pdf).

August 26, 2011—NRC staff provided (for the Commission's approval) SECY-11-0117, "Proposed Charter for the Longer-Term Review of Lessons Learned from the March 11, 2011, Japanese Earthquake and Tsunami."

September 9, 2011—In SECY-11-0124, "Recommended Actions To Be Taken without Delay from the Near-Term Task Force Report," NRC staff presented to the Commission six NTTF recommendations for U.S. nuclear power plants to be initiated without delay.

September 30, 2011—Using Temporary Instruction 2600/0015, "Evaluation of Licensee Strategies for the Prevention and/or Mitigation of Emergencies at Fuel Facilities," NRC resident inspectors began examining U.S. nuclear fuel cycle facilities' plans and procedures for safely dealing with severe events (http://www.nrc.gov/reading-rm/doc-collections/insp-manual/temp-instructions/ti2600-015.pdf).

October 3, 2011—In SECY-11-0137, "Prioritization of Recommended Actions to be Taken in Response to Fukushima Lessons Learned," the NRC staff proposed to the Commission three tiers of prioritization for the Near-Term Task Force recommendations (http://www.nrc.gov/reading-rm/doc-collections/commission/secys/2011/2011-0137scy.pdf).

October 18, 2011—In SRM-SECY-11-0124, the Commission approved the NRC staff's proposal to implement recommendations described in SECY-11-0124 within five years (by 2016).

October 19, 2011—In SRM-SECY-11-0117, the Commission approved the proposed charter with changes.

December 15, 2011—In SRM-SECY-11-0137, the Commission approved the staff's proposed prioritization of the Near-Term Task Force recommendations (http://www.nrc.gov/reading-rm/doc-collections/commission/secys/2011/2011-0137scy.pdf).

January 23, 2012—In SECY-12-0010, "Engagement of Stakeholders Regarding the Events in Japan," the staff updated the Commission on efforts to (1) provide the public with the NRC's most up-to-date understanding of the chronology of events at the Fukushima Dai-ichi Nuclear Power Plant and with the agency's ongoing understanding of the plant's status, (2) obtain stakeholder input on the recommendations provided in the NTTF report and brief the ACRS on the NTTF recommendations and agency plans going forward, and (3) obtain feedback from public citizens on the readability and understandability of the final NTTF report.

February 17, 2012—In SECY-12-0025, "Proposed Orders and Requests for Information in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Tsunami," NRC staff proposed orders in response to lessons learned from Fukushima, provided information concerning the requests for information that the staff planned to send to reactor licensees, and provided a status of the ongoing work conducted under the Charter (http://www.nrc.gov/reading-rm/doc-collections/commission/srm/2012/2012-0025srm.pdf).

March 9, 2012—In SRM-SECY-12-0025, the Commission approved the issuance of the proposed orders subject to changes and comments and, in addition, approved a 50.54(f) letter requesting each reactor operator reevaluate the seismic and flooding hazards at their site using present-day methods and information, conduct walkdowns of their facilities to ensure protection against the hazards in their current design basis, and reevaluate their emergency communications systems and staffing levels (http://www.nrc.gov/reading-rm/doc-collections/commission/srm/2012/2012-0025srm.pdf).

March 12, 2012—The NRC issued orders and the 50.54(f) letter as directed by the Commission in SRM-SECY-12-0025.

- EA-12-050: Reliable Hardened Vents
- EA-12-049: Mitigation Strategies
- EA-12-051: Spent Fuel Pool Instrumentation
- 50.54(f) letters based on the numbered NTTF Recommendations:
  2.1 Seismic Reevaluations
  2.3 Seismic Walkdowns
  2.1 Flooding Reevaluations
  2.3 Flooding Walkdowns
  9.3 Emergency-Preparedness Staffing and Communications

March 20, 2012—The NRC published in the *Federal Register* (FR) an Advance Notice of Proposed Rulemaking requesting comments on specific questions and issues related to addressing SBO conditions.

March 21, 2012—In SRM-SECY-12-0010, the Commission approved, with changes, the staff's plans for ongoing engagement with the public and stakeholders concerning matters related to Fukushima Dai-ichi, the steps being taken to improve communications and outreach, and the continuing effort to leverage existing mechanisms for engaging with other organizations.

April 18, 2012—The NRC published in the *Federal Register* an Advance Notice of Proposed Rulemaking to seek public comments on the potential changes to the Commission's regulations that address onsite emergency-response capabilities for nuclear power plants.

May 31, 2012—The NRC staff issued guidance for U.S. nuclear power plants concerning the assessment of emergency-preparedness communications and staffing, as well as the performance of seismic and flooding walkdowns. Additionally, the NRC staff issued draft guidance for comment for the Orders.

July 13, 2012—In SECY-12-0095, "Fourth 6-Month Status Update on Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Subsequent Tsunami," staff provided the program plans on Tier 3 recommendations and provided the second 6-month status update on Charter activities (http://www.nrc.gov/reading-rm/doc-collections/commission/secys/2012/2012-0095scy.pdf).

August 30, 2012—The NRC staff issued implementation guidance to enable U.S. nuclear power plants to achieve compliance with each of the Orders.

November 26, 2012—In SECY-12-0157, "Consideration of Additional Requirements for Containment Venting Systems for Boiling Water Reactors with Mark I and Mark II Containments," the staff provided the Commission with information, options, and a recommendation to impose new requirements for containment venting systems for BWRs with Mark I and Mark II containments.

January 25, 2013—In COMSECY-13-0002, "Consolidation of Japan Lessons Learned Near-Term Task Force Recommendations 4 and 7 Regulatory Activities," NRC staff requested Commission approval to consolidate regulatory activities associated with NTTF Recommendations 4 and 7 into a single rulemaking.

February 14, 2013—In SECY-13-0020, "Third 6-Month Status Update on Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Subsequent Tsunami," the staff provided the third 6-month status update on Charter activities related to lessons learned from Fukushima.

March 19, 2013—In SRM-SECY-12-0157, the Commission approved Option 2 to issue a modification to Order EA-12-050, "Order Modifying Licenses with Regard to Reliable Hardened Containment Vents," to require licensees of BWRs with Mark I and Mark II containments to upgrade or replace the reliable hardened vents required by Order EA-12-050 with a containment venting system designed and installed to remain functional during severe-accident conditions. The development of technical bases and rulemaking for filtering strategies with drywell filtration and severe-accident management of BWR Mark I and II containments should consider Option 3 (design and installation of an engineered filtered containment venting system intended to prevent the release of significant amounts of radioactive material following the dominant severe-accident sequences at BWRs with Mark I and Mark II containments) and Option 4 (the development of requirements and technical acceptance criteria for confinement strategies and requirements for licensees to justify operator actions and systems or combination of systems, such as suppression pools, containment sprays, and separate filters to accomplish the function and meet the requirements).

June 6, 2013—The NRC issued Revised Order EA-13-109 requiring Severe-Accident-Capable Hardened Vents.

September 6, 2013—In SECY-13-0095, “Fourth 6-Month Status Update on Response to Lessons Learned from Japan’s March 11, 2011, Great Tohoku Earthquake and Subsequent Tsunami,” the staff provided a status update on activities related to Fukushima lessons learned and provided a description of the plans for and status of transitioning oversight of lessons-learned activities from the Steering Committee to the appropriate line organizations, as well as the plan to document closure of lessons-learned activities as they are completed, and requested Commission approval to dissolve the charter to facilitate transfer of lessons learned to the line organizations.

November 15, 2013—The NRC issued Japan Lessons-Learned Project Directorate Interim Staff Guidance (JLD-ISG-)13-02, “Compliance with Order EA-13-109, Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation under Severe Accident Conditions,” for complying with Phase 1 of the Severe-Accident-Capable Hardened Vents Order.

Question No. 31

Concerning the Baseline Risk Index for Initiating Events (BRIIE), could the USA specify the procedure related assigning “a value to each initiating event according to its relative importance to the plant’s overall risk of damage to the reactor core”?

Answer: The relative importance value for each initiating event is calculated using the Birnbaum importance measures. The Birnbaum importance is a measure of sensitivity of core-damage frequency to changes in initiating-event frequency. Birnbaum importance measures for BRIIE are obtained from the standardized plant-analysis risk models for each reactor. An average Birnbaum importance for pressurized-water reactors (PWRs) and BWRs is calculated using the Birnbaum measures of each plant. The process for calculating these importance values is detailed in NUREG/CR-6932, “Baseline Risk Index for Initiating Events (BRIIE).”

Question No. 32

Could the USA provide more details on how the issue of nuclear power plant ageing is managed, especially in order to extend operating duration of the nuclear power plant?

Answer: Plant aging is managed under various programs throughout the life of a nuclear power plant. During the initial license period, plant aging is managed primarily under the Maintenance Rule and other aspects of 10 CFR 50. As stated in section 14.1.4 of the U.S. National Report, “Studies [have] found that facilities deal adequately with many aging effects during the initial license period, and that credit should be given for these existing programs, particularly those under the NRC’s Maintenance Rule, 10 CFR 50.65, ‘Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,’ which helps manage plant aging.”

To extend the operation of a nuclear power plant beyond the initial 40-year license period, an applicant must submit a license-renewal application that includes an integrated plant assessment. As defined in 10 CFR 54.3, an Integrated Plant Assessment “…is a licensee assessment that demonstrates that a nuclear power plant facility’s structures and components requiring aging management review in accordance with 10 CFR 54.21(a) for license renewal have been identified and that the effects of aging on the functionality of such structures and components will be managed to maintain the current licensing basis such that there is an acceptable level of safety during the period of extended operation." Before issuing a renewed license, the NRC staff makes the determination under 10 CFR 54.29 that there is reasonable assurance that the proposed aging-management actions listed in the integrated plant assessment will manage the effects of aging for the identified structures and components during the period of extended operation (PEO).

Question No. 33

The NRC expects to continue to receive additional applications from entities that want to build and operate small and large new nuclear power plants. The NRC pursued the development of small nuclear power plants for several years. Which are the potentially most important safety issues for these small power plants?

Answer: The NRC does expect to receive applications from utilities who do want to build and operate new reactors. To facilitate this safe operation, we have been doing research focused on the following nine key areas, each of which addresses multiple technical topics and activities:

1. framework, including the development of regulatory decisionmaking tools based on risk-informed performance-based principles;
2. accident analysis, including PRA methods and assessments, human factors, and instrumentation and controls;
3. reactor/plant systems analysis, including thermal-fluid dynamics, nuclear analysis, and severe accident and source-term analysis;
4. fuels analysis and testing;
5. materials analysis, including graphite behavior and high-temperature metal performance;
6. structural analysis, including containment/confinement performance and external challenges;
7. consequence analysis, including dose calculations and environmental impact studies;
8. nuclear materials safety (including enrichment, fabrication, and transport) and waste safety (including storage, transport, and disposal); and
9. nuclear safeguards and security.

Some other potential policy, licensing, and key technical issues that the NRC is still in the process of resolving include:

- multi-module risk assessment
- appropriate source term, dose calculations, and siting for small modular reactors
- appropriate requirements for operator staffing for small or multi-module facilities
- security and safeguards requirements for small modular reactors

These lists provide an overview of the safety, technical, and policy issues that the NRC is facing with the application for new light-water small modular reactors.

Question No. 34

The NRC continues to implement Fukushima lessons learned within existing regulatory processes that include review of industry response to orders, requests for information (RFIs), use of operating experience, rulemaking, and conducting additional research.
Which important additional research is under consideration?

Answer: Chapter 11 of NUREG-1925, "Research Activities FY2012–FY2014," (http://pbadupws.nrc.gov/docs/ML1324/ML13242A030.pdf) provides discussion of follow-on research related to Fukushima Dai-ichi, including the following:

- Containment-overpressure mitigation-systems analysis.
- Potential enhancements to the capability to prevent or mitigate seismically induced fires and floods.
- Hydrogen control and mitigation inside containment and other buildings.
- Consequence study of a beyond-design-basis earthquake affecting the spent fuel pool for a U.S. Mark I BWR.
- Fukushima Dai-ichi accident study with MELCOR 2.1.

Question No. 35

What kind of lessons have been learned from the licence renewal projects? What are the main problems during the process?

Answer: The NRC has not encountered any major license-renewal process issues since the inception of the license-renewal program. However, the NRC staff has noted that in instances in which a licensee submits major licensing action requests (e.g., power uprate amendment) concurrently with the license-renewal application, the coordination and explanations of impacts on the current licensing basis might delay the license renewal review because of the additional resources required for such a coordinated review.

Aging-related lessons learned are captured in the GALL Report, the NRC's primary reference document for license renewal. The GALL Report has been revised and updated twice since its original issuance. In between each revision, the staff also captures new lessons learned that might impact aging management through the issuance of ISGs. The latest list of effective ISGs and the issues addressed by them are available at http://www.nrc.gov/reading-rm/doc-collections/isg/license-renewal.html.

Question No. 36

Acceptance of SAFSTOR option - In many countries it is considered the task of the present generation to solve today's waste problems. What is the rationale behind the acceptance of SAFSTOR? What is the reasoning behind the policy to postpone dismantling a NPP?

Answer: "SAFSTOR" is an alternative decommissioning option to DECON (e.g., immediate dismantling and decontamination) that is acceptable to the NRC. Please note that SAFSTOR requires that decommissioning must be completed within 60 years. The rationale for the SAFSTOR option and the related policy for dismantling is to allow radioactive decay to reduce dose rates by approximately a factor of 100 and reduce waste volumes by a factor of 10. (Please refer to Figure 7-4 of NUREG-1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants," and page 20 of Revision 9 of NUREG/BR-0164, "NRC - Independent Regulator of Nuclear Safety," which are available through the NRC public Web site at www.nrc.gov.) In addition, we note that the NRC's regulations requiring monitoring and prior allocation of decommissioning funds using an acceptable financial assurance mechanism would ensure protection of, and minimal financial burden on, future generations. It is also noted that "SAFSTOR" is conceptually identical to the IAEA decommissioning strategy of "deferred dismantlement" described in WS-R-5, "Decommissioning of Facilities Using Radioactive Material Safety Requirements."

Question No. 37

Does NRC have the power to intervene when a plant is operated in compliance with the rules, but an unknown safety issue is identified?

Answer: The NRC has several tools with which to intervene when an unknown safety issue is

identified. Several forms of generic communications are available to notify licensees of important safety information. INs communicate operating or analytical experience to the nuclear industry. Each licensee is expected to review the information for applicability and consider appropriate actions to avoid similar problems at their reactor sites. GLs (1) request licensee actions and/or information to address issues regarding emergent or routine matters of safety, security, safeguards, or environmental significance and (2) require a written response. In addition, Bulletins can be issued to (1) request licensee actions and/or information to address significant issues regarding matters of safety, security, safeguards, or environmental significance that have great urgency and (2) require a written response. If a new safety issue arises that calls into question adequate protection of public health and safety, the NRC could issue orders to modify licenses or require specific actions by licensees. Several orders were issued to licensees following the Fukushima accident.
Question No. 38
Fire Protection Regulation Program - Could you please give more details about lessons learned of experience of plants transitioning to 10CFR50.48(c) (NFPA [National Fire Protection Association] 805) versus those not transitioning, from the point of view of greatest safety improvement?
<u>Answer</u>: 10 CFR 50.48(c) allows plants to voluntarily transition to a risk-informed and performance-based fire-protection program. Plant changes have been proposed as part of the transition that provide significant safety improvements. The determination of the level of safety improvement largely depends on site-specific factors (e.g., level of compliance with the deterministic separation requirements, risk significance of the variations from the separation requirements, and the potential risk benefit of proposed modifications in other risk hazard groups such as internal events, flooding, and seismic events). For example, the Shearon Harris nuclear power plant's transition to 10 CFR 50.48(c) included installing an alternate seal injection system, very-early-warning fire-detection systems, and fire-resistant cables. The Oconee nuclear power plant's transition included the installation of a protected service-water system. These plant modifications significantly improve safe-shutdown capability in the event of a plant fire and also improve safety in the event of other plant hazards (such as station blackout, loss of offsite power, and tornadoes). Not all transitioning plants are proposing this degree of plant changes and plants that are not transitioning are taking steps to resolve deterministic issues such as operator manual actions and spurious actuations, so safety improvement varies based on numerous plant-specific factors.
Question No. 39
What measures does (and can) the NRC employ to raise the standard for those plants that perform in Column 3 and 4? Are there any escalation effect, e.g., if a plant has been in a degraded Column for some periods, does that imply further actions from the NRC?
<u>Answer</u>: The NRC employs a graded regulatory response to licensee performance. The NRC does not change the performance standard that licensees are expected to meet. For plants that enter Columns 3 and 4 of the Action Matrix because of significant performance issues, the agency increases inspection activity for those plants. A Column 3 plant will receive an Inspection Procedure (IP) 95002 inspection, "Inspection for One Degraded Cornerstone or Any Three White Inputs in a Strategic Performance Area," which results in 200 hours of additional inspection. A Column 4 plant will receive an IP 95003 inspection, "Supplemental Inspection for Repetitive Degraded Cornerstones, Multiple Degraded Cornerstones, Multiple Yellow Inputs or One Red Input," which could result in an additional 3000 hours of inspection of all facets of the licensee's programs and organization. Additionally, those licensees are expected to develop a performance-improvement plan and will be invited to publicly brief the Commission on that improvement plan. The regional office will issue a Confirmatory Action Letter to document the

commitments identified by the licensee in its performance improvement plan and any other written or verbal commitments. The Confirmatory Action Letter should explicitly identify licensee actions that, when effectively implemented and validated by the NRC, will provide the necessary assurances that performance improvements are sufficient to transition the plant out of Column 4.

Escalated actions are available for plants in a degraded cornerstone for a lengthy time. If a plant remains in Column 3 for three years or more, that licensee may be invited to a public meeting with the Commission to explain its plan for improving performance. There is a significant incentive for a Column 3 plant to exit Column 3 because another safety-significant finding or Performance Indicator crossing a threshold could force that licensee to move to Column 4, which results in a significant increase in inspection activity and a large expenditure of licensee resources. The nature of the performance issues and the inspection requirements for a Column 4 plant typically cause a plant to remain in Column 4 for several years. Ultimately, if the NRC has concerns that a licensee is unable to improve performance and ensure adequate protection of public health and safety, that licensee can be ordered to shut down a poorly performing reactor unit.

Question No. 40

Who fills in the indicators for the Industry Trends Programs (i.e. who provides the input data, NRC themselves? Industry?)? If the industry provides the input: What oversight or peer check if performed on the input?

Answer: The input data for the Industry Trends Program's indicators is provided by both industry and the NRC. Some of the indicators used in the Industry Trends Program are also used in the ROP's Performance Indicator program. The data for these indicators is voluntarily provided by industry, in accordance with the guidance established in NEI 99-02, "Regulatory Assessment Performance Indicator Guideline." The NRC inspects each plant's reporting of performance-indicator data as part of the baseline inspection program and has inspection procedures in place to collect these data if needed. For other indicators, the NRC collects the data from Event Notifications and Licensee Event Reports, which are reported by industry in accordance with 10 CFR 50.72, "Immediate Notification Requirements for Operating Nuclear Power Reactors," and 50.73, "Licensee Event Report System," respectively.

The data sources for each indicator used in the Industry Trends Program are listed in IMC 0313, "Industry Trends Program." The NRC's processes to collect and verify performance-indicator data are established in IP 71511, "Performance Indicator Verification," and IP 71510, "Discrepant or Unreported Performance Indicator Data."

Question No. 41

Does the predetermined number in the first level (Tier 1 performance monitoring) change/adjust to new safety installations and modernizations (i.e. are the levels dynamic to new plant configurations or new knowledge regarding risks)? If so; who sets these new levels?

Answer: The prediction limits used in Tier 1 of the BRIIE are performance-based. These prediction limits are developed using information from a baseline period, which contains at least four years of data. A statistical test is applied to groups of data, and the period with the most constant frequency is selected as the baseline period. A predictive distribution is calculated for this period, and the value representing 95 percent of the upper limit of initiating event counts in a year is defined as the prediction limit.

Prediction limits are assessed periodically and are revised to ensure these reflect current industry performance; they are not revised to reflect new plant configurations or new risk knowledge and, rather, are assessed and revised based on demonstrated industry performance.

A discussion of how prediction limits for Tier 1 events are calculated is provided in NUREG/CR-6932.

Question No. 42

How is low-frequency events (i.e. events that are not likely to happen at all during the period) treated in this model? Does the NRC count "near-misses" or "minor-degradations" for these types of events instead?

Answer: The BRIIE accounts for nine predetermined types of events for both PWRs and BWRs: loss of offsite power, loss of vital alternating-current (AC) bus, loss of vital direct-current (DC) bus, loss of main feedwater, very small loss-of-coolant accident, general transient, loss of condenser heat sink, stuck-open safety/relief valve, and loss of instrument air. Additionally, BRIIE accounts for steam generator tube ruptures as a tenth initiating event for PWRs. When calculating the Birnbaum importance measures (a ranking of the event's relative contribution to overall risk to plant safety), the frequency for these events is set to either 0 or 1.0 reactor critical year (rcry) to represent whether the event happened or not. If a low-frequency event occurs and results in any of the initiating events monitored in BRIIE, the resulting initiating-event occurrences are accounted for in the model.

These initiating-events indicators are described in NUREG/CR-6932. The staff's recommendation for using these indicators is documented in NUREG-1753, "Risk-Based Performance Indicators: Results of Phase 1 Development."

Question No. 43

How is external events treated in the ASP Program? Have there been any changes due to more extreme weather condition (climate change) and lessons learned from the accident in Fukushima Daicchi NPP

Answer: The treatment of external events in the Accident Sequence Precursor analysis has not changed because of the accident at the Fukushima Dai-ichi plant or because of apparent increase in extreme weather conditions resulting from climate change. However, insights and plant modifications gained from the implementation of Fukushima lessons learned will be included in the Standardized Plant Analysis Risk models as appropriate.

All Accident Sequence Precursor analyses are performed in accordance with Regulatory Issue Summary (RIS) 2006-24, "Revised Review and Transmittal Process For Accident Sequence Precursor Analyses," (http://www.nrc.gov/reading-rm/doc-collections/gen-comm/reg-issues/2006/ri200624.pdf), and the Risk Assessment of Operational Events (RASP) Handbook (http://www.nrc.gov/reactors/operating/oversight/program-documents.html). RIS 2006-24 states that operational events (degraded conditions and/or initiating events) for which there is a Significance Determination Process evaluation do not have a separate Accident Sequence Precursor analysis performed and the Accident Sequence Precursor Program uses the results of the Significance Determination Process evaluations. If no Significance Determination Process evaluation is performed, a separate Accident Sequence Precursor analysis is performed. The RASP handbook provides detailed guidance on performing Accident Sequence Precursor analyses (and Significance Determination Process evaluations).

If an external event or extreme weather condition causes a reactor trip and subsequent loss of mitigation equipment (e.g., offsite power), the Accident Sequence Precursor Program will perform an analysis of the event that occurred. For example, if a tornado causes a loss of offsite power and subsequent reactor trip in which the switchyard is damaged, an analysis is performed for the “loss of offsite power” event similarly to how it would have been done if the loss of offsite power had been caused by an electrical fault not associated with an external event or extreme weather condition. However, events (such as a loss of offsite power) caused by external events or extreme weather typically have higher conditional core-damage probabilities because damage to SSCs precludes recovery of the affected equipment within the PRA mission time (24 hours). The Accident Sequence Precursor analysis will also account for other structures, systems, and components that are adversely affected by the external event or extreme weather. (Note: Accident Sequence Precursor analyses account for any SSC degradation or failure regardless of the cause.)

For degraded conditions with no observed initiating event, the associated external event sequences are treated with nominal initiator frequencies and associated effects on mitigating structures, systems, and components as modeled in the NRC’s Standardized Plant Analysis Risk models.

Question No. 44

What competences is represented in the operating experience evaluating group? Does this group only consist of NRC staff or is there a mix of NRC staff, operators and vendors?

Answer: The NRC’s operating experience group is made up solely of NRC staff with engineering backgrounds. The staff involved has a variety of experience, including operational and maintenance experience in the U.S. nuclear industry and in the U.S. Navy’s nuclear program, as well as previous experience at the NRC in inspection, technical review, licensing review, and incident response. The NRC’s operating experience group interacts regularly, in accordance with a memorandum of agreement, with the industry’s operating experience group at the INPO to exchange information on events of interest and developing trends.

Question No. 45

What role does the industry have in the Reactor Safety Research Program?

Answer: The NRC’s Reactor Safety Research Program is separate and independent from industry. However, the NRC has periodic meetings with both the DOE (tasked with promoting nuclear power) and the EPRI at various levels of the organization. We have existing memoranda of understanding with both organizations. Under these memoranda, we have various addenda related to specific cooperative research programs which help leverage agency resources, obtain access to data and facilities that we might not otherwise have access to, and minimize unnecessary duplication.

Please note that the NRC is always cautious when engaging in cooperative research with EPRI and DOE. We primarily cooperate in the production of needed data. However, we do not collaborate on the evaluation or interpretation of the data.

Question No. 46

How could the USNRC manage the work load to continue the reactor oversight in parallel with the treatment of received reactor licensing applications and current issues, in particular the investigation to draw lessons learned from Fukushima, in a timely manner and adequate quality?

Answer: Each reactor has at least two NRC resident inspectors stationed at the facility to monitor licensee operations and conduct selected baseline inspections. NRC management carefully monitors resident inspector workload to ensure inspectors are able to perform their primary function of ensuring public health and safety. In addition to the resident inspectors, the NRC regularly sends specialty inspectors and inspection teams to reactor facilities to complete the remaining baseline inspections.

Reviews of reactor licensing applications are conducted by staff dedicated to the licensing program and, therefore, they do not impact the allocation of resources to the inspection program. Staff needed to perform activities associated with Fukushima lessons learned was reassigned from other duties throughout the agency to the Japan Lessons-Learned Project Directorate (JLD) and Mitigation Strategies Directorate. These two organizations are dedicated to implement the changes identified from the lessons learned. Those activities are an agency priority. As a result of the impact of Fukushima-related activities on regulatory and licensing reviews, the staff developed and implemented a prioritization process to focus resources on the highest-priority work, which includes the lessons learned from the Fukushima accident. Lower-priority licensing work is completed as resources become available.

Question No. 47

At present there are several investigations on going and further planned. Will the outcome of necessary measures resulting from those investigations be applied as an amendment to in the meantime issued authorizations?

Answer: [Subsequent to the submittal of this question, through email dated January 9, 2014, Mr. Hugo Nilsson (of International Affairs, ENSI) respectfully requested that the NRC not respond to the question. The question was submitted by mistake and should have been withdrawn. Therefore, based on this information, the NRC is not including a response to this question.]

Question No. 48

Approximately how many inspections have been performed by the Regulatory Body in each NPP in respect to the national investigations and actions taken in the light of the Fukushima Daiichi Accident?

Answer: A number of inspections on the following areas have been conducted: emergency equipment, severe accident management, flooding, and seismic.

March 23, 2011—Using Temporary Instruction 2515/183, NRC resident inspectors began reexamining post-9/11 emergency equipment and related items at U.S. nuclear power plants in light of details from the Fukushima accident (http://pbadupws.nrc.gov/docs/ML1107/ML11077A007.pdf).

April 29, 2011—Using Temporary Instruction 2515/18, NRC resident inspectors began examining severe-accident management procedures and training at U.S. nuclear power plants (http://pbadupws.nrc.gov/docs/ML1107/ML11077A007.pdf).

June 27, 2012—Using Temporary Instruction 2515/187 NRC inspectors began reviewing flooding walkdowns at U.S. nuclear power plants. (http://pbadupws.nrc.gov/docs/ML1212/ML12129A108.pdf).

July 6, 2012—Using Temporary Instruction 2515/183 NRC inspectors began reviewing seismic walkdowns at U.S. nuclear power plants (http://pbadupws.nrc.gov/docs/ML1107/ML11077A007.pdf).

Question No. 49

The report describes a Generic Issues Program to address issues that cannot be more appropriately addressed by other regulatory programs or processes. The criteria for those issues appropriate for processing through the program are provided in the report, including those “not being addressed through other regulatory processes or voluntary industry initiatives.

What process is used to identify that an issue is not being otherwise addressed, how is inclusion within the program initiated, and how is it ensured that no issues are missed?
Answer: Anyone, including members of the public, may propose an issue for review by the generic-issues program (additional information is available at http://www.nrc.gov/about-nrc/regulatory/gen-issues.html). Issues that meet the screening process, as described in the CNS report and on the public internet site, are included in the program. While it can never be guaranteed that all important issues are discovered and addressed in the Generic Issues Program, the NRC has committed to take seriously any credible proposal submitted by either an NRC staff member or a member of the public (in the United States or abroad). In addition, NRC management, including program office directors and regional administrators, have defined roles and responsibilities in administering the program, such as forwarding issues for program review that appear to meet the generic-issue guidelines, regardless of how the issue is raised (e.g., through differing professional staff opinion). It is the responsibility of the generic-issues staff to be aware of voluntary nuclear-industry initiatives that might relate to potential issues and, similarly, to know of ongoing agency activities, including research projects that might serve as more appropriate repositories for issues already being addressed.
Question No. 50
The report states that a number of actions were taken in response to the Fukushima accident, and section 16 describes emergency preparedness, including the response of the NRC. However, the report does not appear to describe any requirement for basic plant parameters such as reactor pressure and temperature to be provided outside the plant, or offsite in the case of a severe accident. Does such a requirement exist, and if it does, how is this information propagated to those responding to the emergency?
Answer: Before the accident at Fukushima, the emergency-response data system had been implemented at all U.S. nuclear power plants to provide basic plant parameters to the NRC emergency-response organization as required by the NRC regulation in Appendix E, “Emergency Planning and Preparedness for Production and Utilization Facilities,” to 10 CFR Part 50. The NRC issued guidance (in GL 89-15, “Emergency Response Data System”) to nuclear power plant licensees in 1989; this guidance identified the parameters to be provided to the NRC and also answered questions regarding the implementation of the emergency response data system. This guidance is available at the following Web site: http://www.nrc.gov/reading-rm/doc-collections/gen-comm/gen-letters/1989/gl89015.html. NTTF Recommendation 9.3 to enhance the capabilities of the Emergency Response Data System is discussed in Enclosure 3, “Update on Tier 3 Activities,” to SECY-13-0095, which is available as http://www.nrc.gov/reading-rm/doc-collections/commission/secys/2013/2013-0095scy.pdf. A program plan outlined in Enclosure 3, “Program Plans for Tier 3 Recommendations,” to SECY-12-0095 (http://www.nrc.gov/reading-rm/doc-collections/commission/secys/2012/2012-0095scy.pdf), described an approach to collectively address all Tier 3 Fukushima activities related to emergency preparedness using an Advance Notice of Proposed Rulemaking, a tool that allows the NRC to solicit early written stakeholder input on a new potential rulemaking effort. The NRC plans to take this approach and expects to use the feedback from the advance notice of proposed rulemaking to help determine the need for rulemaking and the scope and content of a proposed rule. The NRC expects to issue the Advance Notice of Proposed Rulemaking in Fiscal Year (FY) 2016.

Question No. 188
Section 6.3.6 describes the Generic Issues Program. Please provide more details for the benefit of other CPs [Contracting Parties] about the "business process improvement techniques" that have been implemented in 2012 to resolve Generic Issues more efficiently.
<u>Answer</u>: The Business Process Improvement approach attempts to systematically evaluate and define a specific process, in this case the generic-issues process. Business Process Improvement specialists team with generic-issues stakeholders and subject-matter experts to review, baseline, and measure current performance. The team then analyzes the underlying issues, identifies root causes, and brainstorms potential solutions. The most effective solutions are then implemented to achieve maximum benefits. The revised process is monitored and data are gathered and reviewed to ensure effective improvements. The following steps may be used by the Generic Issues Business Process Improvement Team: 1. Document the current state of the Generic Issues Program 2. Validate the necessity for each step 3. Brainstorm and categorize Issues 4. Prioritize issues and analyze root causes 5. Brainstorm potential solutions 6. Perform risk analysis on solutions 7. Seek alignment with stakeholders 8. Finalize recommendations and present them to senior management 9. Document the recommendations and implementation plan This structured and systematic approach should help result in a generic-issues program that efficiently initiates, processes, and resolves generic issues. Benefits include: • enhanced stakeholder and subject-matter expert involvement, especially in decisionmaking, that results in good ideas and support; • continuous structured communications with stakeholders that ensure that all aspects of the problem are evaluated and that the solutions developed provide the most effective response to problems; • open discussions among team members that help facilitate critical and creative thinking; • plans that ensure structured implementation in a timely fashion to realize gains as quickly and efficiently as possible; and • clearer delineation of team members' roles and responsibilities.
Question No. 193
According to the 6.3.9 of the 6th National Report, NRC issued the Decommissioning Planning Rule (DPR). Please provide the detailed information on what regulations were changed and why they were changed from the previous regulation on decommissioning of NPPs.
<u>Answer</u>: • The rule added 10 CFR 20.1406(c), which requires licensees to conduct their operations in ways that minimize the introduction of residual radioactivity to the site, including into subsurface soil and ground water. • The rule amended 10 CFR 20.1501, "General [Surveys and Monitoring]," to require licensees to survey residual radioactivity that might be a radiological hazard at the site, including in subsurface areas, and to keep records of surveys of subsurface residual radioactivity identified at the site with records important for decommissioning. • The rule amended financial assurance regulations in 10 CFR Parts 30, 40, 50, 70,

and 72 (for titles refer to http://www.nrc.gov/reading-rm/doc-collections/cfr/) to require materials licensees to (1) incorporate the results in their decommissioning cost estimates and (2) adjust the decommissioning fund as necessary, and to require decommissioning power-reactor licensees to annually report additional information on the costs of decommissioning and spent fuel management.

For more details please refer to SECY-09-0042, "Final Rule: Decommissioning Planning (10 CFR Parts 20, 30, 40, 50, 70, and 72; RIN-3150-AI55)" at http://pbadupws.nrc.gov/docs/ML0905/ML090500143.pdf.

Question No. 194

It is known that there are 4 units of nuclear power plants (Crystal River-3, Kewaunee, San Onofre-2,3) which were permanently shutdowned in 2013. Please provide information on the decommissioning strategies for those reactors and the NRC's regulatory activities for decommissioning of those reactors in 2013.

Answer: Kewaunee and Crystal River Unit 3 have submitted reports indicating they will complete decommissioning in 60 years, which is permitted by our regulations; this strategy is known as SAFSTOR (see the response to question #36). San Onofre, Units 2 and 3, have not yet provided their plans; by regulation, they have two years from their shutdown to submit their plans.

Question No. 195

USA operates Accident Sequence Precursor Program. Explain if there is a new precursor derived after 2010.

Answer: Since FY 2010 (ending September 30, 2010), 41 precursors have been identified; Accident Sequent Precursor Program results are available through FY 2012. A summary of Accident Sequent Precursor Program results is provided to the NRC Commissioners in an annual status paper. The most recent papers are SECY-11-0138 (FY 2010 results), SECY-12-0133 (FY 2011 results), and SECY-13-0107 (FY 2012 results). These SECY papers can be viewed through the NRC public Web site at http://www.nrc.gov/reading-rm/doc-collections/commission/secys/. In addition, the individual Accident Sequent Precursor analysis reports are also publicly available.

ARTICLE 7. LEGISLATIVE AND REGULATORY FRAMEWORK

1. **Each Contracting Party shall establish and maintain a legislative and regulatory framework to govern the safety of nuclear installations.**

2. **The legislative and regulatory framework shall provide for:**

   (i) the establishment of applicable national safety requirements and Regulations

   (ii) a system of licensing with regard to nuclear installations and the prohibition of the operation of a nuclear installation without a license

   (iii) a system of regulatory inspection and assessment of nuclear installations to ascertain compliance with applicable regulations and the terms of licenses

   (iv) the enforcement of applicable regulations and of the terms of licenses, including suspension, modification, or revocation

This section explains the legislative and regulatory framework governing the U.S. nuclear industry. It discusses the provisions of that framework for establishing national safety requirements and regulations and systems for licensing, inspection, and enforcement. It also addresses lessons learned from the Fukushima accident.

Question No. 51
Indicated that all inspection findings are recorded, and the NRC typically issues inspection reports for a specific power plant quarterly. Are those quarterly reports available for public or published on the NRC web site?
Answer: All inspection reports that are publicly available are posted to the NRC public Web site and can be located at the following link: http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/listofrpts_body.html. The Regional Offices are responsible for entering inspection reports into ADAMS and entering information from the reports into the Reactor Program System. Twice each quarter, headquarters staff posts the reports to the public Web site. Inspection reports with security-related information will not be publicly available; however, the cover letter for those reports, which does not include the security information, should be available on the public Web site.
Question No. 52
It is indicated that the United States has not changed the legislative framework governing the US nuclear industry since Fukushima, and that the NRC has taken some regulatory actions in response to Fukushima and continues to consider whether additional actions, such as amendments to NRC regulations, are appropriate (Article 7; Section 2.7.3; page 80). Could you indicate which NRC regulations have been up to now identified as candidates for future amendments in response to Fukushima lessons learned, and what issues should have to be addressed in those amendments?

Answer: It is true that the United States has not changed the legislative framework governing the U.S. nuclear industry since Fukushima. SECY-13-0095 contains the status of all post-Fukushima regulatory activities. Please refer to this document, which is located at http://www.nrc.gov/reading-rm/doc-collections/commission/secys/2013/2013-0095scy.pdf.
Question No. 217
The report states "The NRC must license all commercial nuclear installations (e.g., nuclear power plants) in the United States". Are there any regulatory processes for the export of nuclear installations?
Answer: Section 126 of the AEA authorizes the NRC to regulate and license the export of commercial nuclear installations ("production" or "utilization" facilities in the statute's terms). The NRC established regulatory processes for the export of nuclear installations in 10 CFR Part 110, "Export and Import of Nuclear Equipment and Material." Under both the statute and the NRC's implementing regulations, exports of commercial nuclear installations are strictly regulated and require rigorous preclearance review. For example, both the Commission itself and the Executive Branch must review an application to export a nuclear installation; this cannot be delegated to another, lower-ranking NRC official, even though the Deputy Director of the NRC's Office of International Programs actually signs the license (10 CFR 110.40, "Commission Review," and 10 CFR 110.41, "Executive Branch Review"). Also, there are a host of specific licensing criteria and regulatory prerequisites that must be satisfied before the Commission can issue such an export license. These criteria can be found in 10 CFR 110.42, "Export Licensing Criteria." As an example, the Commission, with input from the Executive Branch, must determine that the export will "not be inimical to the common defense and security" of the United States. For a complete listing of the licensing criteria that apply to these types of exports, please consult both 10 CFR 110.42 and Section 126, "Export Licensing Procedures," of the AEA.
Question No. 229
It is stated in the first paragraph in page 77 that the NRC established the regulations through "notice-and-comment" rulemaking procedures. 1. 10CFR50.55a endorses ASME and IEEE [Institute of Electrical and Electronics Engineers], also Reg. Guide endorses some other IEEE standards. Is there any reason for adopting two ways of endorsement, CFR and Reg. Guide? What are the differences in legal effects when those standards endorsed by CFR and RG are not met? 2. Would you explain the meaning of the "endorsed", "referenced" and "used" when the standards are adopted? 3. Are there any differences in the meaning between "standards" and "consensus standards"? 4. In the rulemaking procedures of 10CFR50.55a, how much manpower and time are required for NRC to review, for example, the ASME sec.III, XI, OM, IEEE-279, IEEE-603 for endorsement?
Answer: 1. The "notice-and-comment" rulemaking procedures referred to on page 77 are derived from the Administrative Procedure Act of 1946, as amended. That statute applies to all

Federal agencies, and it requires that all regulations go through a round of notice and comment before they are effective. What this practically means is that if the NRC wants to issue a new regulation, it first needs to publish that regulation as a "proposed rule" and then allow members of the public to "comment" on that proposed rule. After responding to the comments, the NRC can then issue the rule in final version and it becomes codified as a regulation. Regulations are legally binding on the nuclear industry; failure to comply with a validly enacted regulation (i.e., a regulation that goes through the notice-and-comment procedures) could subject a licensee to enforcement penalties. An RG, on the other hand, does not need to be enacted through notice-and-comment procedures. Even so, the NRC often issues draft RGs for comment to seek public feedback before finalizing the guides. But, unlike a regulation, an RG does not have independent legal effectiveness. So although an RG might offer explanations on how the NRC interprets its regulations, or describe one acceptable way for a licensee to comply with a regulation, it's still the regulation itself that the licensee must comply with or risk facing penalties.

2. The terms "endorsed" and "referenced" are merely ways of pointing out that certain regulatory requirements can be found in external documents. So if the NRC "endorses" or "references" a code in a valid regulation, that code becomes incorporated (so to speak) in the NRC regulation itself.

3. The Society for Standards Professionals provides great information on standards. The following excerpts from its Web site describe standards and consensus standards:

   "Standard—A standard is a document that applies collectively to codes, specifications, recommended practices, classifications, test methods, and guides, which have been prepared by a standards developing organization or group, and published in accordance with established procedures.

   "Consensus standards—Consensus standards are standards developed through the cooperation of all parties who have an interest in participating in the development and/or use of the standards. Consensus requires that all views and objections be considered, and that an effort be made toward their resolution. Consensus implies more than the concept of a simple majority but not necessarily unanimity."

   Additional information on standards can be found at http://www.ses-standards.org/displaycommon.cfm?an=1&subarticlenbr=58.

4. Resources needed (including manpower and time) for rules tend to vary depending on the specific details of each individual rule. However, rulemaking activities for these types of rules require the efforts of perhaps 3 to 5 full-time employees over the course of, on average, 4.5 years.

ARTICLE 8. REGULATORY BODY

1. **Each Contracting Party shall establish or designate a regulatory body entrusted with the implementation of the legislative and regulatory framework referred to in Article 7, and provided with adequate authority, competence, and financial and human resources to fulfill its assigned responsibilities.**

2. **Each Contracting Party shall take the appropriate steps to ensure an effective separation between the functions of the regulatory body and those of any other body or organization concerned with the promotion or utilization of nuclear energy.**

This section explains the establishment of the U.S. regulatory body (i.e., the NRC). It also explains how the functions of the NRC are separate from those of bodies responsible for promoting research, development, and advancement of nuclear energy (e.g., DOE). It discusses the regulatory body's financial and human resources, its international responsibilities, and its policy for maintaining openness and transparency. Lastly, this section addresses lessons learned from Fukushima.

Question No. 53
It is written that NRC don't license DOE's test and research reactors. Are those industrial prototypes or scientific nuclear facilities? Are US scientific nuclear facilities related to commercial nuclear facilities?
Answer: It is true that the NRC does not license DOE's test and research reactors. The U.S. Energy Reorganization Act of 1974 separated the NRC and DOE's predecessor agency into two separate agencies. 10 CFR 50.11, "Exceptions and Exemptions from Licensing Requirements," carefully delineates which facilities are under the NRC's regulations and which are not (namely, government-owned or -controlled DOE sites managed by DOE contract). DOE has several formal policies, orders, and directives that regulate the design, environmental impacts, construction, operation, facility safety, startup, and restart of, and quality-assurance requirements for, all DOE nuclear facilities that ensure that the DOE research test reactors maintain high safety levels. The DOE Office of Health, Safety and Security's Nuclear Safety Research and Development Program provides regulatory oversight for the DOE reactor facilities. DOE currently has two steady-state operating test reactors: (1) the Idaho National Laboratory's Advanced Test Reactor (ATR) and (2) the Oak Ridge National Laboratory's High Flux Isotope Reactor (HFIR). DOE has two transient "pulse"-type test reactors: (1) the Annular Core Research Reactor (ACRR) at Sandia National Laboratories and (2) the Idaho National Laboratory's Transient Reactor Test Facility (TREAT). TREAT is shut down but is under consideration for restart. The Idaho National Laboratory's ATR Critical (ATR-C) reactor facility is a low-power prototype under just-critical low-temperature conditions that mimics the ATR and is used for neutronics benchmarks and detector studies. Sandia National Laboratories also has a critical experimental facility at the ACRR site. More information about these DOE reactors and their critical facilities is available from the following Web page links: • ATR and ATR-C: http://atrnsuf.inl.gov/; a users' guide is available as https://secure.inl.gov/atrproposal/documents/ATRUsersGuide.pdf.

- HFIR: http://neutrons.ornl.gov/hfir/; a users' guide is available at http://atrnsuf.inl.gov/LinkClick.aspx?fileticket=5ru4nSkx7HM%3d&tabid=197.
- ACRR: http://energy.sandia.gov/?page_id=14265 and http://www.sandia.gov/research/facilities/annular_core_research_reactor.html.
- TREAT: Idaho National Laboratory, "Future Transient Testing of Advanced Fuels: Summary of the May 4–5, 2009 Transient Testing Workshop Held at Idaho National Laboratory," INL/EXT-09-16392, September 2009, available as http://www.inl.gov/technicalpublications/Documents/4480294.pdf.

As for non-DOE facilities, U.S. research and test reactors licensed by the NRC are used for research and development purposes. These facilities can be operated for commercial purposes for part of the time.

Question No. 54

In the report is stated that the Subcommittee on Clean Air and Nuclear Safety is responsible for regulation and oversight of the NRC and the Subcommittee on Energy and Power and the Subcommittee on Environment and the Economy have responsibility for regulation and oversight of the NRC as well. Bulgaria would appreciate if USA provides more detailed information about the regulation and oversight carried out by these subcommittees and what are the differences in the oversight they perform?

Answer:

Background
It is important to clarify that the Subcommittee on Clean Air and Nuclear Safety listed above is a subcommittee within the Senate Committee on Environment and Public Works. The other two subcommittees listed above (the Subcommittee on Energy and Power and the Subcommittee on Environment and the Economy) are subcommittees of the House of Representatives Committee on Energy and Commerce. Therefore, both the Senate and the House of Representatives (two very distinct legislative bodies) carry out their Federal governmental oversight responsibilities through their own separate subcommittees.

While the oversight of the NRC falls completely within one subcommittee in the Senate Committee on Environment and Public Works, the House Committee on Energy and Commerce have adopted a slightly different approach. In this approach, programs related to the NRC's regulation of nuclear waste fall within the jurisdiction of the Subcommittee on Environment and the Economy, while everything else related to the NRC falls within the jurisdiction of the Subcommittee on Energy and Power.

Methods of Congressional Oversight
Oversight is a daily activity on Capitol Hill that occurs in various ways. For example, it is conducted through many hearings, in meetings, and even in informal settings such as committee staffer briefings that might not be labeled as "oversight." Overall, the congressional oversight review function is a result of many congressional activities— committee meetings on legislation, the confirmation process, casework, informal member and staff meetings, and legislative communications.

The traditional method of exercising congressional oversight is through committee hearings and asking questions regarding Executive Branch operations. In this regard, both the Senate Committee on Environment and Public Works and the House Committee on Energy and Commerce carry out their responsibilities for oversight of the NRC in a very similar manner. The NRC has been invited to be a witness at a number of hearings over the past few years, as well as having almost daily interactions with numerous staffers of these committees.

Both committees can be considered to be fairly similar when it comes to oversight, because they can both be described as having Committee and Subcommittee Chairs that are committed to doing oversight on a sustained basis; have committee members involved in an activity that might take an extended amount of time and resources; have experienced professional staff with investigatory skills; and are interested in conducting effective follow-through to ensure that any recommendations of the committee are acted on. From this perspective, there is not much difference in the way these two committees conduct their oversight of the NRC.

Question No. 55

In the report is mentioned that the NRC uses an integrated approach to provide new employees with consistent information when it is needed. To assist new employees, the NRC has developed virtual-orientation center. Would you provide more information about this center?

Answer: The Virtual Orientation Center was designed to meet the needs of new NRC employees starting their duty with the Federal government. The Virtual Orientation Center allows new employees to learn more about the agency and provides answers to basic questions related to employee benefits. Links to Commissioner biographies, as well as agency publications such as the NRC strategic plan, provide the new employee with background information on agency leadership as well as the blueprint for the agency. Information on what to expect on the first day of work is also discussed.

Material contained in the Virtual Orientation Center includes:

- Quick facts about the NRC
- Benefit Information (e.g., about pay, leave, health benefits, vision/dental benefits, flexible spending toward health care, and retirement benefits)
- Information about work/life balance (e.g., about the Employee Assistance Program, fitness center, Nursing Mothers Program, childcare subsidy, transportation, and Ergonomic Program)
- Information about reasonable accommodation for individuals with disabilities
- “What to Expect on the First Day” (e.g., orientation information, a new-hire checklist, and information about parking)
- Commission information (e.g., Commissioner biographies, the NRC’s strategic plan, and a welcome letter from the Executive Director for Operations)
- Information about training (e.g., about mandatory and recommended courses for new hires)

Question No.56

Current knowledge management and knowledge transfer activities include the following:
…followed by various activities listed in the report
Please elaborate on those activities, which seem to be suitable to capture best the implicit knowledge?

Answer: Implicit or tacit knowledge is generally difficult to capture. We have found that tacit knowledge is often best captured and transferred in open and informal settings; an unstructured or semi-structured format offers the opportunity for staff to ask questions and learn directly from the-subject matter experts. In most structured formats, experts are often focused on a specific topic or experience; however, when employees have the opportunity to ask questions in a candid setting, experts might recall different experiences, which will result in generating different discussions, thoughts, and questions. As indicated in the initial response, branch and team meetings, "brown-bag" (bring your own) lunches, seminars, and panel discussions all create opportunities to facilitate the transfer and capture of tacit knowledge. Other methods used include interviewing, job shadowing (i.e., watching and learning how a more experienced staff member does their job), coaching/mentoring, and performing rotation assignments (i.e., learning on the job in different assignments). In addition to these methods, it is important to first determine who the experts are, identify the critical skillsets, and then assess the best method for making their knowledge available. We often produce a record of this knowledge, whether it is video, audio, electronic, or on paper.

Question No. 57

The report notes that as of September 30, 2012, the NRC had sufficient funds to meet program needs. Given the fact that the NRC recently had to suspend most of its operations during the US Government shutdown, how has this impacted on the work of the NRC and are there any implications for the manner of provision of NRC's funding in the future?

Answer: The NRC budgets sufficient funds to meet program needs, but did have to suspend most work operations during the last week of the U.S. Government shutdown in October 2013. The annual funding for the U.S. Federal Government expired on September 30, 2013, but the NRC was legally able to continue work operations with limited carryover funds in the absence of an appropriation. For a period of time, employees reported to work as normal. The impact on NRC work operations during this period was that spending on certain discretionary contracts, travel, and training was limited. Eventually, the NRC ran out of carryover funds and had to suspend most of its operations during the U.S. Government shutdown.

During the shutdown, the NRC did not conduct non-emergency reactor licensing, reactor license-renewal amendments, emergency-preparedness exercises, or reviews of design certifications or rulemaking and regulatory guidance. The NRC continued to fund the resident inspectors at nuclear power plants because they are safety-related and are considered "excepted function" employees. The NRC Web site was not updated during the shutdown and routine press releases, meeting notices, plant status, and event reports were not available. The backlog of normally reportable information was posted to the Web site once the shutdown ceased. The NRC continued to receive safety and security concerns through the Web site and hotlines and remained prepared during the furlough to respond to an emergency. The NRC's plan for a shutdown and furlough of employees is contained in Management Directive (MD) 4.5, "Contingency in Lapse of Appropriations," which is available on the NRC's public Web site as http://pbadupws.nrc.gov/docs/ML1015/ML101530672.pdf.

Question No. 58

On retaining staff you mention retaining experienced staff, particularly those who are eligible to retire. Is something done to limit dependence on retirees?

Answer: The NRC has processes in place to ensure that it does not depend on retirees to carry out its functions. For example, the NRC focuses on entry-level hiring through the Nuclear Safety Professional Development Program. Our goal is that 25 percent of all new hires each year are at the entry level. The agency also does a significant amount of hiring at the mid-career level, drawing staff after their military service, from utilities, and from other nuclear fields. This ensures that the agency is creating a pipeline of trained individuals to fill positions as employees retire. If qualified staff is not available to fill a position vacated by a retiree, the agency can rehire the retiree for a limited duration to transfer knowledge and build capacity among the staff. A request to rehire a retiree must be submitted by office management in writing and is presented to a panel of senior agency executives for their review and approval. These hires are limited in duration and a clear exit strategy must be outlined which ensures that the agency has qualified staff ready to assume the position.

Question No. 59
In your report some impressive training results are mentioned like: “FY 2012, 82.2 percent of the 46,004 course completions were conducted as online courses”. From our experience we have concluded that live interactions between students and teachers cater for very useful in-depth discussions. The advantage of in class discussions over one-on-one discussions is that all students present benefit simultaneously. Do these online courses in the USA also include such live interactions?
<u>Answer</u>: Classroom and live synchronous distance learning are important elements of the NRC’s learning and development environment, particularly for qualification-related training of nuclear regulatory and inspection personnel. However, we have found that asynchronous just-in-time-training is also effective for informational and performance-support topics and for certain compliance-related topics. Before determining training-delivery methods, the NRC performs an Instructional Systems Design analysis based on the ADDIE (Analysis, Design, Development, Implementation, and Evaluation) model to determine the appropriateness of training-delivery models. The NRC is working on a blended training-delivery model in which parts of the delivery might be asynchronous and parts might be instructor-led.
Question No. 60
Pilot project with video interviews with retiring staff – is this an initiative that will be continued? Are there already conclusions about its effectiveness, and if so, what are these?
<u>Answer</u>: Video- and audio-recording techniques are still evolving. We’ve received feedback from users about the length, searchability, and reuse of videos. Employees often want to view only a segment or a certain topic in a video that might be very long. This might require the user to watch the video in its entirety just to find the information for which they are looking. As a lesson learned, we have begun to preplan the format and structure of these videos, taking the overall topic and creating subtopics to separate into video segments. As a result, we can provide the user a pinpointed time or location in the video for certain areas of a topic, providing the user the information in a more efficient manner. We are currently looking into techniques and tools to support categorization and archiving of media files that might make it possible to navigate and search videos.
Question No. 61
The Subsection describes results of an independent international peer review of the safe operation issues, i.e. the IAEA’s OSART mission at Seabrook Station Unit 1. "One of the identified areas for improvement is for the plant to be more proactive in resolving long-term issues". Please, clarify.
<u>Answer</u>: Issues identified relative to this recommendation include (1) low-level groundwater seepage affecting different areas of the plant since shortly after construction; (2) the existence of significant backlogs (corrective-action, work-order, and procedure-change requests); and (3) the high average age of backlog items. The Seabrook followup OSART mission was conducted in June 2013. Concerning these issues, the followup mission found that Seabrook has revised its corrective-action process, taken multiple actions to resolve a number of long-term issues, and initiated a cultural change to enhance staff awareness. The followup mission concluded that satisfactory progress has been made to date.
Question No. 62
There are financial constraints for the U.S.NRC because of the sequester since March 2013. Even more severe cuts might be possible in the upcoming months following the government shutdown etc. The NRC stated at the RIC [Regulatory Information Conference] in March 2013

that cuts will affect research at a disproportionate rate. This may cause severe trouble not only to the U.S., but also to international research programs and projects like the Halden Reactor Project. When continued over a longer term, such a development may compromise safety goals, e.g. in fuel safety, where new fuel and cladding types are used with higher burnups and safety research has to catch up with these developments.

"Will the NRC continue to cut the spendings for reasearch at a disproportionate rate, also for even more severe cuts as caused by the sequester? Does the NRC dispose of a concept to decide at which point more cuts compromise nuclear safety, especially when looking at research? Has the NRC established a ""red line"" for this?"

Answer: Sequestration will continue to result in a decreased level of effort in a variety of research activities; in particular, research projects that require longer-term developmental work and for which the impact won't be immediately realized. Our strategy to address these significant reductions includes:

- bringing a significant portion of the work in house,
- cutting back or shedding lower-priority research,
- continuing current (and looking for new) cooperative research programs,
- strategically assessing additional opportunities to leverage technical work performed by international counterparts, and
- encouraging DOE and the nuclear industry to perform research necessary to confirm industry proposals.

We are involved in numerous cooperative research programs both domestically and internationally that help us leverage agency resources, obtain access to data and facilities that we might not otherwise have access to, and minimize unnecessary duplication. We are continuing to support most of these programs during this time of limited budgets and funding uncertainty. We are also looking for additional opportunities to leverage our research funding as we move forward. Many organizations are similarly facing funding challenges resulting in cooperative research programs being not only attractive but necessary.

The agency will ensure that the sequestration cuts do not affect research activities that would impact the NRC's ability to perform the critical safety and security oversight for existing licensees.

Question No. 63

According to the report, the NRC coordinates with INPO to implement the hosting of an OSART mission in the United States every 3 years. The last one was conducted at Seabrook Station Unit 1 in 2011, the next is planned to take place at Clinton Power Station, Unit 1, in 2014. Projected on the great number of NPPs in the U.S., this would mean that every plant benefits from an OSART review every about 300 years. This is not much.

Considering the developments after the Fukushima accident, is there the will of the NRC to change the OSART policy and host more missions in the coming years?

Answer: The U.S. has long supported the OSART program and continues to do so. The NRC stresses the importance of peer reviews to its licensees and encourages (but cannot compel) their participation in the OSART program.

Additionally, the U.S. nuclear industry has a review program through INPO, which is an non-governmental organization formed by the U.S. nuclear industry following the Three Mile Island accident. INPO conducts plant evaluations, provides industry training, performs event analyses and information exchanges, and provides assistance to nuclear plants on an as-requested basis. While INPO's activities do not replace the NRC's oversight function, INPO does provide the industry with a means of self-assessment that complements the NRC's role. INPO performs onsite plant evaluations approximately once every two years for every U.S. operating nuclear power plant. INPO evaluations are based on U.S. requirements and practices. It should also be noted that once every six years, each INPO evaluation is conducted as a WANO peer review and includes at least four international peers.

OSART missions are based on IAEA Standards, which are similar to NRC regulations. The U.S. has been hosting an OSART mission once every three years and participating in OSART missions in other countries annually. The U.S. believes that this level of onsite engagement is appropriate as part of assuring harmonization with international operational standards.

Question No. 64

The Fukushima accident has shown that design and siting, especially the back-fitting of plants according to the state of the art of science and technology, are crucial topics. Taking into account the developments since the CNS extraordinary meeting in 2012 , are the United States in favor of extending the scope of OSART missions from mainly operational issues to design and siting issues?

Answer: The U.S. believes that the current scope of OSART missions is properly focused on plant operations and should not be extended to design and siting issues. However, Recommendation 2.2 of the NRC's NTTF report (http://pbadupws.nrc.gov/docs/ML1118/ML111861807.pdf) suggests a periodic update of the reevaluated hazards (e.g., seismic and flooding) based on any new and significant information since the most recent reevaluation. Recommendation 2.2 is prioritized as Tier 3, which is used to address items that (1) require further NRC staff study to support a regulatory action, (2) have an associated shorter-term action that needs to be completed to inform the long-term action, or (3) depend on the availability of critical skill sets. In this case, the NRC's assessment of possible periodic updates of external hazards will be informed by the results of the reevaluations of seismic and flooding hazards that are currently underway as part of NTTF Recommendation 2.1. Because those reevaluations are not complete, the NRC has not yet assessed or developed possible regulatory requirements for the periodic reevaluations of hazards usually considered as part of the siting and design phase of nuclear power-plant licensing and construction. When sufficient insights are gained from the seismic and flooding reevaluations, and if a periodic reevaluation of external hazards is deemed appropriate to consider, the NRC would use the standard rulemaking process, which includes developing a technical basis and engaging stakeholders for public participation, to address such concerns. This is a years-long process that will consider the appropriate scope and timing of such reevaluations, which would in turn determine how they might be addressed in domestic and international activities, including OSART missions conducted in the United States.

Question No. 65

Some States have shown a desire to participate in matters relating to nuclear power plants. States are authorized only to observe and assist in NRC inspections of reactors, they cannot conduct their own independent health and safety inspections. What form of assistance do State representatives offer in NRC inspections of reactors?

Answer: NRC's MD 5.2, "Cooperation with States at Commercial Nuclear Power Plants and other Nuclear Production or Utilization Facilities," contains specific and separate guidance for States to request to participate as an observer or active participant in an NRC inspection.

In terms of observation, the NRC allows States to observe NRC inspection activities and inspection entrance and exit meetings. MD 5.2 describes the specific process to be followed (see pages 7 through 9), including signing a protocol agreement. States are allowed to observe all types of inspections (routine, special, reactive, etc.) and inspections of specialized areas such health physics, security, and emergency preparedness.

Some examples of recent State observations of NRC inspections include:

- reactor-site baseline inspections and reactor-site infrequent inspections (e.g., of reactor-head and steam-generator replacement)
- force-on-force inspections
- groundwater inspections
- license-renewal inspections

In addition, over a longer period, State counterparts also have accompanied the NRC during inspections at:

- fuel cycle facilities
- complex decommissioning sites
- reactor decommissioning sites

In terms of State personnel as active participants in an NRC inspection, MD 5.2 describes the process to be followed (see pages 9 and 10). At the current time, Illinois is the only State which has entered into a formal Memorandum of Understanding with the NRC to be allowed to participate in NRC inspections. State of Illinois personnel have participated in many types of NRC planned and reactive inspections, such as:

- reactor-site baseline inspections and reactor-site followup inspections
- force-on-force inspections
- groundwater inspections
- license-renewal inspections
- reactor-site infrequent inspections (e.g., of reactor-head and steam-generator replacement)

It should be noted that some States might have cooperative agreements with licensees that allow State personnel presence onsite. These agreements are done through outreach initiatives by licensees and are arrangements solely between the State and the licensee, with no NRC involvement. The States are respectful of the NRC's jurisdiction and ensure that appropriate issues are brought to the agency's attention when this is needed.

Question No. 66

It is noted that the NRC has arrangements to exchange technical information with other foreign regulatory authorities. Please could you indicate how the exchange of technical information is coordinated so that information is shared in a timely and efficient manner, and so that the NRC is able to improve its own technical oversight through the internal promulgation of information.

Answer: The NRC's Office of International Programs is responsible for coordinating technical cooperation between the NRC and its foreign regulatory counterparts. The Office of International Programs serves as a liaison between our foreign counterparts and the technical staff within the NRC's various program offices. Within the Office of International Programs, the staff within the International Cooperation and Assistance Branch is assigned a portfolio of countries for which they serve as the primary point of contact for technical cooperation. Similarly, staff within the Export Controls and International Organizations Branch is responsible for coordinating the NRC's technical cooperation with international organizations such as the IAEA and the Nuclear Energy Agency. When commitments are made by the NRC or by its foreign counterparts during technical exchanges, the Office of International Programs is responsible for tracking those commitments to ensure that they are appropriately addressed.

NRC senior managers meet periodically as part of the NRC's International Council. The International Council sets goals and priorities for international activities and provides a forum for managers to exchange information, address questions, and solve problems. Additionally, each of the technical program offices within the NRC has one or more staff members that serve as international liaisons. These liaisons meet regularly with the Office of International Programs staff as part of the NRC's International Council Working Group to ensure that the NRC's international cooperation and assistance is appropriately coordinated internally between the policy staff within the Office of International Programs and the technical staff within the program offices.

Question No. 67

The NRC has been subjected to a decreased budget in 2012 (page 96), and yet has a need to recruit and retain technically capable staff to offset losses and enhance its capability. This involves a range of activities including a "grow your own" program for PRA specialists. To what extent are funding limitations affecting the ability of the NRC to recruit and retain staff. Are there other areas with similar issues and if so what is being done to address them?

Answer: While the NRC has faced budget constraints, as have other agencies in the Federal government, we have been able to recruit new staff to replace those lost through attrition (primarily retirements) and to address critical skill areas. The NRC maintains a focus on entry-level hiring through the Nuclear Safety Professional Development Program and generally ensures that about 25 percent of all new hires each year are at the entry level. The NRC also has a robust campus outreach program, which includes grants for scholarships, fellowships, faculty development, and curriculum development to help ensure the continued conduct of research, development, and training activities in nuclear-related fields of study. Students who receive an NRC grant-funded scholarship or fellowship incur a service obligation to work at the NRC or in the nuclear industry for a period after graduation. The agency also has an onsite presence on campuses through participation in career fairs, student information sessions, and guest lectures. The agency University Champions support the majority of these events. University Champions are senior-level staff at the agency who serve as emissaries of the NRC and establish a close individual liaison with school officials. The NRC is also able to attract staff at the mid-career level, drawing them as they leave military service, and also recruits from utilities and other nuclear fields. Job stability, work/life balance, and Federal retirement benefits are key to attracting mid-career professionals to the NRC.

Another option that the NRC has used to build skills in critical areas is through the Graduate Fellowship Program. The Graduate Fellowship Program sends current NRC employees to universities across the United States to obtain advanced degrees (Masters and Doctorates) in those critical skill areas. Such areas have included human factors, PRA, and health physics, among others. The NRC employees complete a 2- or 3-year program of study, earn an advanced degree, and are obligated to return to work at the NRC for a certain period of time depending on their course of study.

Question No. 68

The report describes plans to retain staff, including those that are eligible to retire. Whilst this may retain skills for a short time, the knowledge transfer program should be an increasing proportion of the potential retirees time. How is knowledge management actively managed and prioritised to ensure that maximum benefit is gained from this?

Answer: The NRC engages in executive succession planning to identify potential successors for executive positions. Succession planning helps guide executive development and informs staffing decisions. An executive board comprising senior leaders periodically meets to determine development that would benefit executives and prepare them for such NRC positions; the board considers strategies for filling positions for which NRC has few potential successors. As positions are determined and executives are placed, the appointee is given sufficient time for knowledge transfer and transition. Knowledge management is not an additional requirement outside normal work scope, but should be integrated into day-to-day operations. In order to meet each organization's unique culture and technology needs, it is important that knowledge management be embedded in each organization. In the past, NRC offices have identified their areas of highest knowledge vulnerability by asking themselves whether they were about to lose the only person in the organization with specific knowledge or experience and whether specific areas would require targeted recruiting. Recognizing that "one size does not fit all," NRC offices were empowered to select and implement the best strategy for their particular situation. Many have developed individual strategies and plans for knowledge management for their particular office. The knowledge-management program advocates a number of techniques and tools for knowledge management that are mentioned in the report and which many offices adopt and use.
Question No. 69
In the light of the previous questions, will it be necessary, or is it planned, that technical support organisations will be used to provide support for regulatory activities, and if it is, how is it intended that these will be managed to avoid the risks of loss of regulatory oversight?
Answer: The NRC strives to maintain within its staff core expertise in all technical disciplines needed to fulfill its regulatory obligations. For the technical experience and knowledge that does reside within the staff, continuous efforts are made to recruit and develop new staff and to retain the knowledge of senior staff through various knowledge-management activities. New staff members are paired with subject-matter experts within the staff of the NRC or of the U.S. national laboratories for on-the-job knowledge transfer. Additionally, training and development plans are employed for standardized and systematic knowledge acquisition. The NRC has begun a series of knowledge-management documents (whose document numbers start with "NUREG/KM-") in which retired staff or senior staff document the important history of regulatory activities. On occasion, emergent issues might demand specific expertise that does not reside within the staff. In addition, variability in workloads requires augmentation of staff expertise. In these cases, the NRC uses contracted experts to augment NRC staff expertise. In all cases, the NRC defines the scope of work, maintains technical oversight of the work, and retains the responsibility for regulatory decisionmaking.
Question No. 70
What kind of experience do you have related using social media? What are the main advantages and disadvantages of using it?
Answer: The NRC has been using social media since it launched a blog on WordPress.com in January 2011. Since that time, the NRC has added Twitter, YouTube, and Flickr to its social media program. The addition of Facebook in mid-2014 is being pursued. As with any new technology, social media have benefits and drawbacks. The primary benefit of social media is the ability for regulators to reach out and talk directly to the public—and hear back from the public—without going through a third party. However, social media can also lead

to a quick and impossible-to-stop spread of misinformation and they require significant resources to use successfully.

It should be noted that social media do not replace traditional means of communicating with the public. There remains a need for press releases, fact sheets and even public meetings to communicate with stakeholders, the public, and the media. But because a growing number of people communicate exclusively through social media, the NRC believes their use is no longer just an optional endeavor. In our experience, we have discovered that social media platforms:

1. Give regulators a new, additional channel for information distribution;
2. Can reach new audiences, especially in certain demographic groups that might not access information in traditional ways;
3. Allow for synergy between different platforms and agency Web sites;
4. Allow for maximum reach through forwarding, sharing, etc., of social media content;
5. Promote discussions and dialogue;
6. Allow regulators to hear what the public is saying and provide a timely feedback loop that allows tailoring of messaging or information to meet the identified need;
7. Allow quick dissemination and repetition of messages during a crisis.

Social media is not just a vehicle for disseminating information. It is also a means for listening to the public. By monitoring social media, particularly Twitter, and reading comments posted to the NRC blog, the NRC can obtain real-time feedback about the success or failure of messages or communication endeavors and can create additional communication products as necessary.

Question No. 201

How does NRC establish and maintain a strong nuclear safety culture for itself? How does NRC establish an internal management system and keep it effective continuously?

Answer: The NRC chartered an Internal Safety Culture Task Force in October 2008 to consider potential initiatives that could improve the agency’s internal safety culture. The Task Force identified several high-level themes for continuous focus and improvement. The recommendations developed by the Task Force address the themes and are focused on creating effective and lasting improvements that will support a strong safety culture for the agency. Activities to support those recommendations include: creating and filling a job position with assigned duties to focus on the agency’s internal safety culture, developing and issuing a matrix on the NRC intranet that allows employees to select the most appropriate avenue in which to raise a concern, and developing training to address interpersonal skills necessary to support the agency’s safety culture position. In addition, the NRC views the tenets of the NRC’s Safety Culture Policy Statement, which was issued in July 2011 and is applicable to all licensees and certificate holders, as having been incorporated in the Agency’s activities reinforcing the NRC’s Mission and Values.

Since 2011, the agency has embarked on an effort to more seamlessly integrate internal safety culture with the overall organizational culture. To assist in that effort, the results of the Office of the Inspector General’s Safety Culture and Climate Survey (SCCS), conducted triennially and administered by a third party, enable the agency to closely examine participants’ responses to questions addressing internal safety culture.

This administration of the SCCS and the actions taken to address the results provide the agency with a vehicle for continuous improvement. The 2009 SCCS drove the creation of the well-received training programs in civility, crucial conversations, and emotional intelligence;

in 2012, the themes for agencywide action were performance management, employee development, valuing human differences, the environment for raising concerns, and evaluation of the Differing Professional Opinions, Non-Concurrence Process, and Open Door Policy programs. In combination with the Office- and Region-level and Division-level actions, numerous initiatives for improvement are in progress.

In addition, to incorporate the theme of "valuing human differences"—an area highlighted in the SCCS in which performance was high, but had fallen from the 2009 survey— the Office of the Executive Director of Operations sponsored a program entitled "Behavior Matters" which enables any and all NRC employees to participate in the identification and propagation of behaviors that align with and illustrate not only the NRC's values of Openness and Collaboration, but also of Integrity, Service, Cooperation, Excellence, and Respect (ISOCCER). By engaging employees at all levels in this dialog, ownership and accountability are shared regarding how interactions are to be conducted across the NRC. Approximately 900 (approximately 25 percent of the total population) staff, managers, and senior leaders participated in this initiative. Phase II will focus on skill-building around self-awareness, situational awareness, and self-regulation.

Management System Components/Programs
It should be noted that under the Whistleblower Protection Enhancement Act of 2012, Federal employees are protected from retaliatory action resulting from filing a complaint regarding "a violation of a law, rule or regulation; gross mismanagement; gross waste of funds; an abuse of authority; or a substantial and specific danger to public health or safety."

The NRC expands on the federally guaranteed rights afforded to federal workers with several processes available to employees for expressing and having their mission-related concerns and differing views heard and considered by management, including the Open Door Policy (included in MD 10.160, "Open Door Policy"), the Non-Concurrence Process (included in MD 10.158, "NRC Non-Concurrence Process"), and the Differing Professional Opinions Program (included in MD 10.159, "The NRC Differing Professional Opinions Program"). The NRC is actively engaged in assessing these processes to determine whether the processes are operating as intended and to identify potential areas of improvement. Data-gathering activities include formative evaluation from employee feedback, results from the triennial SCCS, external benchmarking, targeted surveys, feedback on revised Non-Concurrence Program and Differing Professional Opinion Program guidance, record reviews, and an audit conducted by the Office of the Inspector General. Because of the nature of the processes and the importance of employee engagement and support for the processes, all employees were encouraged to comment on the MDs for the processes. Collectively, this information will support future revisions of the MDs.

In addition to the formal programs mentioned above, the NRC promotes a positive safety culture, with a focus on an open, collaborative work environment. An open, collaborative work environment is an environment in which all employees and contractors are encouraged to promptly raise concerns and voice differing views without fear of retaliation. The NRC has taken several actions to reinforce the importance of this environment, including communications from management, employee seminars, supervisor meetings, and agency newsletter articles. Open, collaborative work environment is promoted on the home page of the NRC's internal Web site.

Finally, as another component of the management system supporting a positive safety culture at the NRC, the agency supports NRC Team Player Awards, which symbolize the value of considering varied approaches and opinions in the decisionmaking process. Recipients are nominated by fellow employees and presented with the award from the Executive Director for Operations; their success stories are included in an agency newsletter.

References:

- Whistleblower Protection Enhancement Act of 2012: http://www.gpo.gov/fdsys/pkg/PLAW-112publ199/pdf/PLAW-112publ199.pdf
- Open Door Policy: http://pbadupws.nrc.gov/docs/ML0414/ML041490186.pdf
- Non-Concurrence Process: http://pbadupws.nrc.gov/docs/ML1317/ML13176A371.pdf
- Differing Professional Opinions: http://pbadupws.nrc.gov/docs/ML0417/ML041770431.pdf
- Management Directives: http://www.nrc.gov/reading-rm/doc-collections/#man

Question No. 202

Page 30 “cumulative effects of regulation” -- please share NRC how to handle with cumulative effects of regulation and current status in USA?

Answer: Since 2010, the NRC has implemented several rulemaking-process changes to curb the cumulative effects of regulation, including:

1. publishing draft guidance with proposed rules and final guidance with final rules;
2. requesting specific comment on cumulative effects of regulations when the proposed rule is published;
3. holding a public meeting on implementation during the final rule stage; and
4. increasing interaction during all phases of rulemaking.

Currently, the NRC staff is responding to further Commission direction on cumulative effects of regulation, including considering whether to expand the process to reduce the cumulative effects of regulation beyond rulemaking, developing a template for identifying the cumulative effects of regulation, and working with the industry to explore the accuracy of costs and schedule estimates within the NRC’s regulatory analyses.

Question No. 230

Which office(s) under the NRC responsible for technical support (i.e. an in-house TSO [technical-support organization])? What are the level of external TSOs' (e.g. National Laboratories, universities, private contractors, etc.) involvement in the NRC¡¯s process of regulatory decision making? If there is some involvement of external TSOs, what are the mechanisms and policies implemented by the NRC to avoid the conflict-of-interest with the nuclear-promotion organizations, as the external TSO could also make contract with them?

Answer: The NRC program offices employ individuals with the technical expertise needed to support the NRC’s mission. Some examples of these program offices include the Office of New Reactors (NRO), the Office of Nuclear Reactor Regulation (NRR), the Office of Nuclear Materials Safety and Safeguards (NMSS), and the Office of Nuclear Regulatory Research (RES). These offices work with technical experts at external entities on an as-needed basis; for example, the Office of New Reactors might work with a national laboratory to help prepare its environmental analysis.

Although the NRC may engage experts at national laboratories, which are owned by the DOE but operated by contractors, all final official regulatory decisions are made by the NRC. So all work products prepared by contractors on behalf of the NRC are reviewed by an NRC employee with technical expertise at the NRC. Making regulatory decisions is an inherently governmental function that cannot be, and is not, delegated to these labs.

Because DOE labs are owned by DOE but operated by contractors, the NRC does review labs for organizational conflict of interest (this is a standard practice for any U.S. government contracting). For example, the NRC does not place work with DOE labs that do work for the nuclear industry or have industry executives involved in their management. This is how the NRC ensures that it avoids conflicts with nuclear-promotion organizations.

Question No. 231

It is mentioned in the National Report that "Ultimately, however, the NRC must make the decision and accept responsibility for it". What would be the responsibility-related consequences for the NRC if a regulatory decision by the NRC is deemed wrong or unjust by the Congress?

Answer: As an agency within the executive branch of government, all NRC regulatory decisions are subject to varying degrees of Congressional oversight. If Congress believes that an NRC regulatory decision is "wrong or unjust," it has various tools that it can use to try to remedy the situation.

The first way Congress can respond if the NRC makes a regulatory decision that is deemed "wrong" would be to enact a new law that modifies or reverses the NRC's decision. But in order to do this, both houses of Congress would need to approve the law and the President would need to sign it. It is important to note that in the context of adjudications, Congress cannot second-guess the NRC's decision because that would affect the due-process rights of specific persons or license holders. Congress can only change the licensing requirements that would apply to future NRC licensing actions.

In addition to actually reversing an NRC regulatory decision through enacting a new law, Congress can also hold oversight hearings. At these hearings, a specific Congressional committee can call both the NRC Commissioners and agency employees and have them testify under oath about the NRC's regulatory practices. Usually, Congress invites the Commissioners themselves to testify. But Congress has the authority to call any NRC employee. At these oversight hearings, Congress can voice its displeasure about certain regulatory decisions or seek answers as to why the NRC adopted a certain course of action. But Congress cannot probe the minds of the Commissioners in any pending adjudicatory matter. Nor can Congress change the NRC's regulatory decision unless it follows the procedures for enacting a new law.

All in all, the NRC enjoys open and professional relations with our oversight committees, who recognize the valuable and objective technical expertise that the NRC brings to bear when making its regulatory decisions. The NRC also recognizes the valid role that Congressional oversight plays in the U.S. constitutional structure.

Question No. 232

It is stated in the National Report that "The NRC has also continued its work both with the IAEA and on a bilateral basis in support of countries seeking to develop new nuclear power programs or expand small or dormant programs". Since several US-based and US-related nuclear corporations are also promoting their technologies in these countries, does the NRC have any official policy to separate its regulatory support activities from such promotion-related activities by American companies?

Answer: The NRC provides international assistance to countries that, for the most part, have requested the NRC's help in strengthening their regulatory programs. To avoid duplication of efforts made by the IAEA, the NRC often establishes assistance projects after considering projects already underway through the IAEA's Department of Technical Cooperation. However,

the regulatory assistance provided to recipient countries is technology-neutral and is not coordinated with the nuclear industry or with any promotional activities conducted by other agencies within the U.S. Government.
Question No. 238
What staff training on severe accidents and resident inspector training on severe accident management guidelines is planned to be provided as a result of the Fukushima lessons learned? What will be the scope of this training? A reference where more detailed information is already available on this issue would be welcome.
<u>Answer</u>: Development of staff training on severe accidents and resident inspector training on severe accident management guidelines is a Tier 3 activity. This activity is discussed in Enclosure 3 to SECY-13-0095 (http://www.nrc.gov/reading-rm/doc-collections/commission/secys/2013/2013-0095scy.pdf). This lessons-learned activity originated from NTTF Recommendation 12.2 to enhance NRC staff training on severe accidents, including resident inspector training on SAMGs. Because the Emergency Onsite Response Capabilities rulemaking (Tier 1) is expected to require better integration of emergency procedures, including SAMGs, this activity partially depends on the final outcome of that rulemaking activity. However, the NRC staff is working toward implementing several potential enhancements related to severe-accident training: • Increasing the frequency of severe-accident courses, including exporting the courses to the NRC regional offices; • Updating courses with lessons learned from the Fukushima accident; • Modifying existing qualification programs to include requirements for severe-accident courses; • Adding SAMG courses to qualification-program training; • Developing additional new courses that focus on severe accidents; and • Establishing and developing new qualification plans for the users of codes that model the consequences of severe accidents. NRC staff recognizes that additional changes could be developed as a result of the ongoing State of the Art Reactor Consequence Analysis study, the Level 3 PRA study, and any future Fukushima lessons-learned insights. The ongoing activities are leveraging existing processes to evaluate and modify training programs. This includes mechanisms for stakeholder communication, where appropriate. The staff does not anticipate any significant technical or policy issues with regard to training enhancements.

ARTICLE 9. RESPONSIBILITY OF THE LICENSE HOLDER

Each Contracting Party shall ensure that prime responsibility for the safety of a nuclear installation rests with the holder of the relevant license and shall take the appropriate steps to ensure that each such license holder meets its responsibility.

The NRC, through the AEA, ensures that the primary responsibility for the safety of a nuclear installation rests with the licensee. Steps that the NRC takes to ensure that each licensee meets its primary responsibility include the licensing process, discussed in Articles 18 and 19; the ROP, discussed in Article 6; and the enforcement program, discussed below. This section provides an update on the licensee's responsibility for maintaining openness and transparency and a discussion of lessons learned from Fukushima.

Question No. 71
Could a parent company be the owner of a license holder? If that is the case: To what extent is that parent company allowed to affect daily operations? Is this regulated?
Answer: Yes, a parent company can be the owner of a license holder without an NRC license. Following deregulation of the electric industry in the United States, some NRC licensees developed contractual arrangements with non-owner operators. In some cases, the owner of a nuclear power plant or nuclear facility holds a license to possess the facility and a separate entity holds the license to operate the facility. In each case, each NRC licensee is authorized to possess, use, and/or operate the facility in accordance with the terms and conditions of their license. Any entity that possesses or uses NRC-regulated materials must be issued a license by the NRC and is ultimately responsible to comply with NRC regulations to ensure safe operation of the facility. As stated in RIS 2001-06, "Criteria for Triggering a Review under 10 CFR 50.80 for Non-Power Operator Service Companies": "It is difficult to identify precisely the point at which an operating service entity must be added to the operating license and thus when an application for NRC approval must be filed. Clearly, some areas are more important than others, and each operating agreement the NRC reviews is likely to be unique. The more operational areas in which an operating entity has final decision-making authority, the more likely it is that NRC must review and approve an application for a license transfer and an amendment to add the operating entity to the license."
Question No. 72
The report states that the NRC may hold a predecisional enforcement conference or a regulatory conference with a licensee before making an enforcement decision. The purpose of the conference is to obtain information to assist the NRC in determining whether an enforcement action is necessary and, if so, what the appropriate enforcement action is. The conference focuses on areas such as common understanding of facts, root causes etc. Are predecisional enforcement conferences or regulatory conferences also oversight instruments that are applied in the course of an event investigations? If yes, would you please outline their utilization in this context (e.g. objective, point in time of the investigation process, etc.)?
Answer: When escalated enforcement action appears to be warranted (i.e., Severity Level I, II, or III violations, civil penalties or orders), a predecisional enforcement conference may be conducted with a licensee before the NRC makes an enforcement decision. Regulatory conferences are conducted in lieu of predecisional enforcement conferences if violations are

associated with risk-significant findings evaluated through the Significance-Determination Process (i.e., Red, Yellow, or White findings). Conferences are held after the NRC has conducted its preliminary inspection/investigation and a preliminary enforcement strategy has been formulated. Conferences are normally held within 60 days of the NRC notifying the licensee that the NRC is considering escalated enforcement action for an apparent violation (or violations) of its regulatory requirements. The purposes of the conference are (1) to provide a forum for the licensee to present new information or perspectives that have been obtained following completion of the initial NRC inspection/investigation that might warrant reconsideration of the preliminary enforcement strategy for the case and (2) to give the NRC an opportunity to assess the reasonableness of the licensee's corrective action. Based on the information presented at the predecisional enforcement conference or regulatory conference, the NRC may (1) issue a Notice of Violation (with or without a monetary Civil Penalty), (2) issue an Order, or (3) take no enforcement action. A predecisional enforcement conference or regulatory conference is normally the final step in the NRC's fact-finding process before making an enforcement decision. The conference is viewed as a means for the licensee to provide the NRC information it believes the agency should consider in determining the appropriate enforcement action and significance determination and is not a meeting to negotiate sanctions with the staff.

Question No. 73

The report states the NRC offers its licensees the opportunity to participate in the Alternative Dispute Resolution Program. Alternative dispute resolution is a general term encompassing various techniques for resolving conflicts outside a court using a neutral third party.

What is the proportion of "alternative dispute resolutions" compared to the annual number of NRC enforcement actions? I.e. is alternative dispute resolution a common method for the NRC and its licensees to help clarify safety issues? If yes, what are the reasons (historical, cultural, etc.) for choosing this approach to help parties clarify issues.

<u>Answer</u>: The NRC implemented its enforcement alternative dispute-resolution program in 2004 to resolve discrimination and other wrongdoing cases. Since that time, the number of cases involving licensees and contractors that have elected to participate in the program has averaged 6 cases per year, which accounts for approximately 5 percent of the overall number of escalated enforcement actions issued.

Either the alternative dispute resolution or the more traditional approach to resolving enforcement issues provides adequate protection of public health and safety and common defense and security. The NRC's use of alternative dispute resolution has resulted in improved public safety from the resulting settlement agreements by including broader and more comprehensive corrective actions than might have been achieved through the traditional enforcement process. The settlement agreements are enforced through confirmatory orders that are publicly available. The opportunity for the staff and the other party to communicate openly with the assistance of a trained mediator helps the staff reach effective agreements that further the NRC's interests. For example, through alternative dispute resolution, the NRC has been able to obtain fleetwide corrective actions at operating reactor facilities where it otherwise could not through the traditional means.

Alternative dispute resolution is a voluntary process and any party may withdraw from the negotiation at any time. It is not used for parties to deliberate the facts of the case, but to reach a settlement agreement. Parties involved have commented that alternative dispute resolution is a less confrontational means to resolve issues than the traditional enforcement process, largely because of the improved communication.

Question No. 218
The report states “The NRC has three primary enforcement sanctions available: notices of violation, civil penalties, and orders”. Can you identify which enforcement tool needs to get approval from the Commission?
Answer: The NRC Enforcement Policy is a Commission document that provides direction to the staff on how the Agency’s Enforcement Program is to be implemented. It is a public document available in ADAMS through Accession Number ML13228A199 and at http://www.nrc.gov/about-nrc/regulatory/enforcement/enforce-pol.html. Issuing enforcement actions in accordance with the direction specified in the Enforcement Policy requires no prior Commission approval. However, certain enforcement actions require either advance written notification to the Commission or advance consultation with and approval by the Commission, depending on the nature of the proposed sanction. Written notification is provided to the Commission for specific escalated enforcement cases, including all enforcement actions involving civil penalties or Orders. Specific enforcement actions requiring prior Commission approval include, but are not limited to, (1) proposals to impose a civil penalty in excess of three times the Severity Level I base value for a particular licensee type, (2) any enforcement action that involves a Severity Level I violation, (3) any proposal to use discretion to impose a daily civil penalty, or (4) an enforcement action affecting a licensee’s operation that requires balancing the public health and safety or common defense and security implications of not operating against the potential radiological or other hazards associated with continued operation. Section 2.3.10 of the Enforcement Policy provides more detail about when Commission notification or prior approval is required.

ARTICLE 10. PRIORITY TO SAFETY

Each Contracting Party shall take the appropriate steps to ensure that all organizations engaged in activities directly related to nuclear installations shall establish policies that give due priority to nuclear safety.

NRC policies that give due priority to safety covered under this article are PRA policy statements and policies that apply to licensee safety culture and safety culture at the NRC. Other articles (e.g., Articles 6, 14, 18, and 19) also discuss activities undertaken to achieve nuclear safety at nuclear installations. Updates to this section discuss lessons learned from Fukushima.

Question No. 74
The report states that for new reactors licensed under 10 CFR Part 52, the NRC requires applicants to describe the design-specific PRA and its results for a design certification application and a plant-specific PRA and its results for a combined license application (Article 10; Section 10.3; page 114). In addition, the NRC requires the holder of a combined license to develop a Level 1 and a Level 2 PRA before initial fuel load and it must cover those initiating events and modes for which NRC-endorsed consensus standards on PRA exist. Each holder of a combined license must maintain and update the PRA every 4 years with upgraded consensus standards that exist at that time until operations permanently cease. Finally, before any application for license renewal, a combined license holder must upgrade the PRA to cover all modes and all initiating events. We would appreciate if you can inform us about the following: a) What kind of scope is required for the PRA study before the "initial fuel load"? b) The scope of PRA study is only for "Internal Events" or includes other internal and external risks of the facility? c) Are considered in the scope the different operation modes, such as low power reactor state, shutdown reactor state, etc.? d) What scope is required to the periodical updating of PRA study every four years?
<u>Answer</u>: The PRA study “before initial fuel load” must include a Level 1 and Level 2 analysis and must treat those initiating events and operating modes covered by NRC-endorsed consensus standards in existence one year before the fuel load date. Currently, that would include the following for full-power operation only (a standard for low-power operation and shutdown does not currently exist): • internal events (e.g., loss-of-coolant accidents, transients, station blackout, anticipated transients without scram, etc.) • fires and floods initiated by sources inside the plant (internal fire and flood) • fires and floods and any other hazard initiated by sources outside the plant (external fire and flood) • high winds (e.g., from a tornado or hurricane) • seismic events A PRA that is upgraded every four years must have a scope that is consistent with NRC-endorsed consensus standards in existence at least one year before the date the upgrade begins. Therefore, at a minimum, the scope would include those initiating events and operational modes listed above, because they have already been endorsed by the NRC. If a standard for

low-power and shutdown operational modes is endorsed by the NRC at least one year before a PRA upgrade, that upgraded PRA would be required to address accident sequences during low-power and shutdown modes of operation initiated by the initiating events covered in that standard.
Question No. 75
Do you use the PSA for evaluation of the NPP life extension after the project lifetime? Can you shortly describe how do you use it?
Answer: The use of probabilistic safety assessment (PSA, or PRA as it is commonly known in the United States) is not a requirement for the renewal of plant operating licenses issued under 10 CFR Part 50 or renewal of a combined license issued under 10 CFR Part 52 because the Commission believes that the methodology for conducting an integrated plant assessment needed to ensure the appropriate management of SSCs' aging should be deterministic in nature. Nevertheless, the NRC recognizes that a plant-specific PRA can be used as an effective tool to provide integrated insights into the plant design, resulting in an additional relative measure of overall plant safety. In addition, the Commission also acknowledges that PRA can be an effective tool to provide added assurance that all SSCs important to license renewal have been appropriately evaluated.
Question No. 76
For licensees with significant performance degradation, the US NRC conducts an independent assessment of the licensee's safety culture through a specific additional inspection procedure. Could the USA give more explanations about this independent assessment? Is it conducted by inspectors, does it include human factor specialists? What kinds of data are collected: documents, interviews with licensee' staff, observation of work situations, etc.?
Answer: Licensees with degraded performance are assessed on a graded inspection approach. Once a licensee has moved into the "multiple/repetitive degraded cornerstone" column of the NRC's action matrix, the NRC will request that they perform a third-party independent safety-culture assessment. The NRC's own safety-culture assessment will review that third-party assessment and make a determination as to the scope of the NRC's assessment. If we have confidence in the methodology used by the licensee's third-party vendor, our own assessment might be narrower in scope. The NRC's assessment is conducted by experienced staff with educational backgrounds ranging from human factors engineering to sociology/psychology and/or extensive experience with safety and organizational culture. In addition, the NRC has a qualification training program specifically designed to qualify employees as Safety Culture Assessors. During the NRC's safety culture assessment, the staff will review documents pertaining to safety culture and safety-conscious work environment policy and procedures, training, and communications to the site's employees. The NRC will conduct focus groups with licensee personnel and individual interviews with members of the licensee's management team. The NRC will also conduct behavioral observations of work being performed in the field and in the control room.
Question No. 77
The NRC plans to update all guidance and inspection documents appropriately with the new common safety culture language in 2013. Please provide further clarification on the new common safety culture language.
Answer: The safety culture common language has been incorporated in the ROP through the revised IMC 0310, "Aspects Within the Cross-Cutting Areas," and is applicable to inspections beginning after January 1, 2014. Conforming changes will be made to other ROP documents.

Question No. 78
Reintegration of Security into the Reactor Oversight Process Assessment Program," dated March 14, 2012, this reintegration became effective on July 1, 2012. This reestablished the original framework of the Reactor Oversight Process, which involved a holistic assessment of licensee performance. This reintegration of the security issues into the Reactor Oversight Process is considered as a good practice.
Answer: Thanks for acknowledging this. We appreciate your comment.
Question No. 79
How does the regulatory body train its inspectors in the inspection of safety culture?
Answer: The NRC currently has elements of safety culture and safety-conscious work environment embedded within the Inspector Qualification program under IMC 1245, "Qualification Program for Operating Reactor Programs." Several training courses discuss the ROP's relationship with safety culture and the cross-cutting components. In addition, the NRC has developed a safety-culture assessor qualification card which is specific to qualifying individuals who do safety culture assessments at the plants. This qualification card is more detailed then the initial inspector training in safety culture and is designed to help typical inspectors become subject-matter experts on how to conduct safety-culture assessments. Completing the qualification card requires attending classroom and Web-based training courses, completing individual study activities, and receiving on-the-job training. The qualification process also requires individuals to participate in an interview with NRC management and qualified Safety Culture Assessors to confirm that the individual has the necessary knowledge, skills, and abilities to independently conduct safety-culture assessments.
Question No. 80
What role does deterministic safety analysis have for the assessment? (The focus seems to be on PRA and Risk-informed.) How is postulated events treated within this model/view?
Answer: In developing a PRA, analysts consider a wide range of hazards and resulting initiating events. Deterministic safety analyses often provide a convenient starting point for developing PRA success criteria. However, one goal of a PRA is to be as realistic as practicable; therefore, design-basis calculations might be too conservative for use in a PRA. In such cases, realistic calculations may be performed to justify the success criteria or to predict accident-sequence progression. In its approach to risk-informed regulation, the NRC blends traditional engineering approaches with risk insights in making regulatory decisions. In addition to considering PRA results, the risk-informed process requires consideration of defense in depth and safety margins. For changes to the licensing basis of a U.S. nuclear power plant, the risk-informed approach requires that performance-monitoring strategies be employed to ensure that the change does not produce unintended consequences. In this way, the NRC's risk-informed approach incorporates insights from both the deterministic safety analyses and the PRA.
Question No. 81
Since the US does not perform periodic safety reviews, i[s] any other method used to achi[e]ve the purpose of a periodic safety review? Are US considering to implement periodic safety reviews in the future?
Answer: The NRC does not require licensees to perform periodic safety reviews at predefined intervals. Instead, the NRC has established processes to ensure that licensees perform continuous review and maintenance of safety of their facilities and their licensing bases. The

licensing basis for nuclear power plants is established on issuance of the license and evolves throughout the term of the operating license because of the continuing regulatory activities of the NRC as well as the activities of the licensee. Licensees implement quality-assurance program requirements (in accordance with Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” to 10 CFR Part 50) and the Maintenance Rule (10 CFR 50.65) for active structures and components. Licensees are also required to perform assessments of modifications under 10 CFR 50.59, “Changes, Tests and Experiments,” and submit license amendments under 10 CFR 50.90, “Application for Amendment of License, Construction Permit, or Early Site Permit.” In addition, the NRC evaluates licensee operating experience; information from inspections, audits, and investigations; and regulatory research. As necessary, the NRC requires changes to the licensing basis for licensees through the release of new or revised regulations, the issuance of orders modifying licenses, and acceptance of licensee commitments to modify nuclear power-plant designs and procedures (e.g., in response to licensee events or generic communications). In such cases, the NRC follows established processes that ensure that the appropriate NRC actions (e.g., rulemaking, the hearing process, and backfit analysis) are taken with full consideration of the safety significance of the issue and with opportunity for stakeholder involvement. Finally, the NRC requires implementation of AMPs for passive components as part of licensees’ preparation of, and the NRC’s review and issuance of, license-renewal applications.

In response to a suggestion by the IAEA Integrated Regulatory Review Service (IRRS) mission in 2010, the NRC is also conducting a pilot study to examine assessment reports for selected periodic safety reviews from other regulators to assess for potential insights regarding nuclear power-plant operating-experience topics. The goal is to glean potential insights that might inform the NRC’s regulatory process and add value to the examination of the example periodic safety reviews.

Question No. 82

Many new studies of level 3 PRA consider the social and socioeconomic impact of a nuclear accident. Will these aspects be taken into consideration for the upcoming level 3 PRA?

Answer: On September 21, 2011, the Commission directed the NRC staff to proceed with a scope-comprehensive site level 3 PRA.

Option 3 of SECY-11-0089, “Options for Proceeding with Future Level 3 Probabilistic Risk Assessment Activities,” describes the approved PRA methodology, which includes the Standardized Plant Analysis Risk (SPAR) model. SPAR models use a standard set of event trees for each plant design class and standardized input data for initiating event frequencies, equipment performance, and human performance (although these input data may be modified to be more plant- and event-specific when needed).

This Level 3 PRA includes Level 2 PRA analyses and also models the transport and dispersion of released radioactive materials to estimate various measurements of offsite radiological-health and economic consequences.

SECY-11-0089 can be found at the following link:
http://www.nrc.gov/reading-rm/doc-collections/commission/secys/2011/2011-0089scy.pdf

Question No. 83

The report states that NRC Internal Safety Culture Task Force developed a set of actions which are targeted at improving the commissions own safety culture. Actions include the appointment of an agency Safety Culture Program Manager.

Would you please outline – if already specified – the function and task description of this position?
<u>Answer</u>: The Safety Culture Program Manager serves as the lead and spokesperson for the Agency's internal safety-culture activities. The Program Manager leads and coordinates efforts to develop, implement, and maintain policies and a framework for a strong internal safety culture by working with senior managers, managers, and staff across the agency to establish and foster implementing strategies to establish a culture of safety that is integrated into the daily activities of the agency. The Program Manager also functions as an advocate for internal safety-culture activities in the agency by conducting and coordinating significant activities to monitor and strengthen the internal safety culture, including performing an advisory role for related initiatives by others (e.g., offices, groups, and management) as appropriate, and assessing activities and programs directed at fostering a strong safety culture. The Safety Culture Program Manager performs these duties, among others: • Identify and recommend policies and procedures directed at establishing and maintaining a strong internal safety culture. Identify and work with appropriate staff to implement changes to the strategic plan and agency performance-management tools in order to establish and maintain a strong safety culture. • Identify and work with appropriate offices to develop and oversee agency programmatic guidance to administer a strong safety culture. Identify, develop, and work with appropriate personnel to develop and conduct training for management and staff so that they can gain an understanding of and foster a strong safety culture. • Serve as an agency advocate to promote awareness of safety-culture concepts by working with offices to establish office-level actions, programs, or processes to clearly integrate safety-culture thinking into their operations. Assist employees in selecting the most effective avenue for addressing safety-culture issues and, when appropriate, raise issues on behalf of others. Identify and develop communications necessary to promulgate the avenues available for employees to raise concerns. Assist offices in efforts to address staff perceptions that raising differing views could have a negative impact on an employee's career or work environment. • Lead efforts to analyze, assess, and evaluate agency programs to ensure that those programs targeted at establishing and maintaining a strong safety culture are effective and achieve the desired objectives; recommend improvements to programs when necessary. Work with appropriate staff to establish programs to monitor the health of the agency's internal safety culture and work with senior management, line management, and staff across the agency to implement lasting improvements when necessary. • Coordinate activities with agency staff engaged in improving the safety culture of the licensee to ensure the alignment (where applicable) and consistency of guidance. • Analyze program evaluation results and interpret and consolidate complex and diverse views and recommendations into concise positions. Prepare and present results of analyses and studies to management. Develop a periodic report addressing the agency's efforts to establish a culture of safety in order to measure the effectiveness of various activities and identify areas needing additional management focus.

- Conduct short- and long-term planning to assist in policymaking and program operations. Represent the office, when assigned, to speak on policy issues. Develop issues, prioritizations, analysis, resolution plans, and programs, as assigned, in their area of responsibility. Be responsible for closely monitoring the financial expenditures of contractors under direct cognizance and recommend budget adjustments or justifications as appropriate.

Question No. 203

What are the main approaches by which the NRC has to takes to ensure the effectiveness of the safety management guidelines?

Answer: From an overall perspective, the NRC reviews the license application of each plant to ensure compliance with regulations that provide reasonable assurance of adequate protection of public health and safety. Changes to a plant's licensing basis must be made in accordance with change processes prescribed by regulation (e.g., 10 CFR 50.59 and 10 CFR 50.90). The NRC has an extensive oversight program, which includes onsite inspectors, to assess a licensee's compliance with the regulations.

The NRC maintains awareness of new information and methods applicable to nuclear power plants and monitors the operating experience of these facilities, both those licensed in the United States and worldwide. The NRC has a number of well-established processes to assess new information or issues and take any necessary regulatory action.

GL 88-20 required plants to search for potential severe-accident vulnerabilities, both from internal and external hazards. All currently licensed nuclear power plants in the United States performed a PRA for the internal-events portion of this search. Other risk-assessment approaches were applied to external hazards, although a number of plants performed seismic or fire PRA studies. Risk insights from the GL 88-20 responses were compiled into two NUREG reports.

NRC inspectors use risk insights to focus their inspection activities in such a way that inspection resources are more efficiently used.

For reactors licensed under 10 CFR Part 52, the NRC requires applicants to describe the design-specific PRA and its results for a design certification application and a plant-specific PRA and its results for a combined license application. In addition, the NRC requires the holder of a combined license to develop a Level 1 and a Level 2 PRA by initial fuel load, which must cover those initiating events and modes for which NRC-endorsed consensus standards on PRA exist. Each holder of a combined license must maintain and upgrade the PRA every 4 years until operations permanently cease. The upgraded PRA must cover NRC-endorsed consensus standards on PRA in effect one year before each required upgrade. Finally, before any application for license renewal, a combined license holder must upgrade the PRA to cover all modes and all initiating events.

Question No. 204

Would you please provide some examples of the interfaces that could affect nuclear safety or security in this section?

Answer: NRC RG 5.74, "Managing the Safety/Security Interface," describes a method for U.S. licensees to assess and maintain changes to safety and security activities.

The following activities are examples of the safety/security interface:

1. Erection of delay barriers by the security organization to impede movement without considering the need for the operator to move freely during an accident could adversely

impact safety.
2. Taking safety-related equipment out of service for maintenance or repair without informing the security organization might undermine the effectiveness of the physical security protective strategy because the security force might dedicate responders to protecting out-of-service equipment.
3. Upgrading safety-related plant equipment to incorporate digital technology without involving the security organization might introduce cyber vulnerabilities which go unaddressed.

Question No. 205

RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," what specific requirements have been added in the Revision 2 compare to the first one? How to implement and supervise that Risk-Informed Special Treatment?

Answer: Revision 2 to RG 1.174 did not include any major changes in NRC regulatory positions for risk-informed changes to a plant's licensing basis. The revision made editorial changes and changed the overall format to conform to the general format for NRC RGs. References regarding technical adequacy of a PRA model were added to RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," and references regarding uncertainty were added to NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making."

The risk-informed special treatment of SSCs is permitted by 10 CFR 50.69. Guidance for licensees who wish to implement this regulation may be found in RG 1.201, "Guidelines For Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance." RG 1.201 endorses, with limitations and clarifications, NEI guidance document NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," dated July 2005.

An applicant for or holder of a license to operate a light-water reactor issued under 10 CFR Part 50 who wishes to implement 10 CFR 50.69 must respectively (1) submit an application to the NRC or (2) request a license amendment from the NRC; this request undergoes a safety review by the NRC staff. An applicant for design approval, a combined license, or a manufacturing license issued under Part 52 may voluntarily address the requirements in 10 CFR 50.69 in their application. 10 CFR 50.69 requires that licensees and applicants update their Final Safety Analysis Report to reflect which systems have been categorized using the risk-informed process. 10 CFR 50.69 includes change-control requirements to ensure proper implementation of risk-informed categorization over time. The licensee's categorization is subject to oversight by NRC inspectors.

Question No. 206

What is the application status of risk monitor at plants?

Answer: All U.S. nuclear power plants have "internal events at power" PRA models, including models for internal flooding hazards. Many plants have developed "risk monitors" (e.g., using the R&R Workstation "EOOS" program) to aid in determining configuration-specific risk; that is, the increased risk when one or more SSCs is out of service for maintenance. While all U.S. nuclear power plants are required by 10 CFR 50.65 to assess and monitor the risk of proposed maintenance activities, there is no requirement that a PRA model be used for this purpose. While many plants use a "risk monitor" for this assessment, the NRC does not keep track of whether a given plant uses such a tool because there is no requirement for its use.

One U.S. plant has been approved to implement risk-managed technical specification completion times. This requires that configuration-specific risk be assessed using a PRA model of sufficient scope, level of detail, and technical adequacy. While a risk monitor would be one method of performing this assessment, that licensee does not use a risk monitor, but rather uses a lookup table that contains risk assessments of various combinations of SSCs out of service (on the order of many thousands of plant configurations).

ARTICLE 11. FINANCIAL AND HUMAN RESOURCES

1. Each Contracting Party shall take the appropriate steps to ensure that adequate financial resources are available to support the safety of each nuclear installation throughout its life.

2. Each Contracting Party shall take the appropriate steps to ensure that sufficient numbers of qualified staff with appropriate education, training, and retraining are available for all safety-related activities in or for each nuclear installation, throughout its life.

This section explains the requirements about financial resources that licensees must have to support the nuclear installation throughout its life, and the regulatory requirements for qualifying, training, and retraining personnel. It also discusses lessons learned from Fukushima.

Question No. 84
The USA mentions control room simulator training for operating crew. Are these simulators also used for people who work closely or interact frequently with operating crew in control room (tests, safety engineers, etc.)? Are there simulators available for training US NRC inspectors or experts? Which kind of specific simulators exist for training and requalification of people such as maintenance staff? Are they available for training contractors?
<u>Answer</u>: The NRC inspects licensees' use of simulators to ensure compliance with current regulations. For example, plant-referenced simulators are used to conduct operating tests as required by 10 CFR 55.45(a), to conduct requalification training as required by 10 CFR 55.59(c)(3), and to perform control manipulations that affect reactivity to establish eligibility for an operator's license as required by 10 CFR 55.31(a)(5). The NRC has anecdotal information that licensees use simulators for other purposes, such as to conduct emergency drills and to train the operating crew before special evolutions. Unfortunately, the NRC does not have specific information as to the extent of how licensees use their simulator for purposes that are outside NRC requirements. The NRC's technical training center uses simulators to train inspectors, operator licensing examiners, and other technical experts. The NRC trains on the following types of control-room simulators: General Electric BWR; Westinghouse PWR; Combustion Engineering PWR; and Babcock & Wilcox PWR.
Question No. 85
The report states that NRC has noticed increased examples of non-conservative decisions that facility licensee personnel have made over the past few years, and the NRC has provided additional inspector guidance when reviewing certain decision that licensees have made. Would you please outline the guidance the NRC provided to its inspectors to turn their attention to non-conservative decision making?
<u>Answer</u>: From 2010 through 2011, the NRC noted an increase in the number of events driven by non-conservative decisionmaking by licensee personnel. This increase was noted both in the more significant events that led to NRC reactive inspections and in lower-level issues that were identified as inspection findings with very low safety significance. IP 71111.11, "Licensed Operator Requalification Program and Licensed Operator Performance," was revised in 2011 to

add four hours per quarter for resident inspectors to observe operators in the control room. The revision was intended, in part, to have the inspectors become familiar with licensee “conduct of operations” procedures before observing control-room activities and to have the inspectors observe control-room operators during periods of heightened activity or high risk. The inspection procedure was revised to provide guidance for inspectors to use in evaluating operator performance in the plant and in the main control room with respect to, among other things, procedure usage, interpretation and diagnosis of plant conditions, use of human-error prevention techniques, and conservative decisionmaking.

In addition, the NRC issued Operating Experience Smart Sample 2012-02, “Technical Specification Interpretation and Operability Determination,” to provide guidance for inspector review of licensee operability determinations and technical specifications. The smart sample provides examples of non-conservative interpretations that were made in the past when licensees failed to properly evaluate technical specification requirements, resulting in violations of regulatory requirements. Also, additional inspection resources were provided for the review of licensee operability determinations under IP 71111.15, “Operability Determinations and Functionality Assessments,” to provide inspectors with the time necessary to ensure that conservative decisions are being made with respect to the operability of safety-related equipment.

ARTICLE 12. HUMAN FACTORS

Each Contracting Party shall take the appropriate steps to ensure that the capabilities and limitations of human performance are taken into account throughout the life of a nuclear installation.

This section explains the NRC program on human performance. This program has seven major areas: (1) human-factors engineering issues, (2) emergency operating procedures and plant procedures, (3) working hours and staffing, (4) fitness for duty, (5) the Human Factors Information System, (6) support for event investigations and for-cause inspections, and (7) training. This section also discusses lessons learned from Fukushima.

Question No. 86
Was the estimation of influence for human factor in communications and subordinations performed taking into account Fukushima lessons both for regulatory system and for operators?
Answer: We understand this question to be an inquiry regarding NRC actions taken in response to Fukushima lessons learned pertaining to the influence of human factors on emergency communications and decisionmaking authority. Following the accident at the Fukushima Dai-ichi nuclear power plant, the NRC established the NTTF to conduct a systematic and methodical review to determine whether it should make additional improvements to its regulatory system. The NTTF's recommendations, which are available in "Recommendations for Enhancing Reactor Safety in the 21st Century", addressed both emergency communications and decisionmaking authority. NTTF Recommendation 8 concerned strengthening and integrating onsite emergency-response capabilities such as emergency operating procedures and severe-accident management guidelines and contained extensive damage-mitigation guidelines; one portion was a specific sub-recommendation to require licensees to specify clear command and control strategies for their implementation. The NRC staff is presently addressing this recommendation by developing a proposed amendment to NRC regulations. NTTF Recommendation 9 concerned requirements for facility emergency plans to address station blackout and multiunit events, including several more specific recommendations pertaining to communications capabilities. The staff requested from the NRC licensees an assessment of their ability to maintain communications throughout the postulated event. Licensees established interim actions using portable satellite phone radios until they install permanent equipment such as new radio systems, sound-powered telephones, battery-powered repeaters, and satellite phones.
Question No. 87
How does the USA intend to ensure that lessons on human and organizational factors learned from the Fukushima Daiichi NPP accident are taken into account in improvement and assessment of the nuclear facility safety, including studies for a better understanding of decision making and the performance of actions in severe unanticipated situations, and the role of contractors in such situations?
Answer: As licensees implement orders and new regulations related to Fukushima lessons learned, they will be inspected in accordance with the requirements of the ROP. Human and organizational factors are cross-cutting areas of the ROP, which are fundamental performance attributes that extend across all of the ROP cornerstones of safety. These areas are human performance, problem identification and resolution, and safety-conscious work environment. These areas are assessed in accordance with the requirements of NRC IMC 305, "Operating Reactor Assessment Program."

For more information on the ROP, please visit http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/. For more information on IMC 305 and the operating reactor assessment program, please refer to http://pbadupws.nrc.gov/docs/ML1317/ML13178A032.pdf.
Question No. 88
Are license holders required to have expertise on human factors in house?
Answer: No. The NRC's regulations do not require licensees to directly employ human factors experts. However, Revision 3 of NUREG-0711, "Human Factors Engineering Program Review Model," provides guidance to the NRC staff concerning the review of licensee human-factors engineering programs, including specific guidance on human-factors engineering program management and composition of the human-factors engineering design team (in Section 2 and the Appendix of the NUREG, respectively). As noted in the Appendix, there is no intent to prescribe any particular organizational structure for an applicant, nor is it assumed that human-factors engineering is the responsibility of a single organizational unit called "the human-factors engineering design team." As such, licensees might be able to meet NUREG-0711 guidelines without in-house human-factors experts (e.g., through the use of contractor experts).
Question No. 89
How does the regulatory body assure that the operators/license holder continuously strengthen nuclear safety from a systemic perspective including human, technological and organizational factors?
Answer: The ROP, which includes performance indicator thresholds and the Significance Determination Process for inspection findings, is used to measure reactor licensee performance and take appropriate regulatory action. The ROP is a risk-informed, performance-based, tiered approach to assessing plant safety. There are three key strategic performance areas: reactor safety, radiation safety, and safeguards. Within each strategic performance area there are cornerstones that reflect the essential safety aspects of facility operation: initiating events, mitigating systems, barrier integrity, emergency preparedness, public radiation safety, occupational radiation safety, and security. Certain aspects of licensee performance are common to all the cornerstones and contribute to maintaining safe facility operation. These aspects are commonly referred to as cross-cutting areas and include human performance, the establishment of a safety-conscious work environment, and problem identification and resolution. Licensee deficiencies in these cross-cutting areas generally manifest themselves as the root causes of performance issues in the cornerstones. The NRC reviews licensee problem-identification and -resolution programs as part of baseline inspections and during a biennial team inspection. The establishment of a safety-conscious work environment is monitored throughout the year by the NRC resident staff through review of allegations and as part of the problem-identification and -resolution biennial team inspection. While there is no specific NRC inspection for human performance, human performance is reviewed as part of a number of baseline inspections and is implicit in the data reported for many of the performance indicators. Although licensees meeting ROP performance-indicator and inspection thresholds are not evaluated specifically for continuous improvement, licensee performance that fails to meet ROP thresholds triggers increased inspections, which can include specific technical inspections as well as corrective action and safety-culture inspections. NRC IPs guide inspectors in these areas. As an example, NRC IP 95003 includes inspection objectives and guidance for (1) independently assessing the adequacy of programs and processes used by the licensee to identify, evaluate, and correct performance issues and

(2) evaluating a licensee's third-party safety-culture assessment and conducting a graded assessment of the licensee's safety culture. In addition, the NRC has issued a Safety Culture Policy Statement (http://www.gpo.gov/fdsys/pkg/FR-2011-06-14/pdf/2011-14656.pdf), which identifies nine attributes of a positive safety culture, including continuous learning.

Question No. 90

The report states that the Human Factors Information System is designed to store, retrieve, etc. human performance information extracted from NRC inspection and licensee event reports.

Is there a link between the Human Factors Information System and the Monitoring of Licensee Safety Culture (see chapter 10.4.1)?

Answer: The staff does not presently use the Human Factors Information System (HFIS) as a means to monitor licensee safety culture.

Question No. 91

The report mentions the field of activity of human factors experts.
What are the technical competences needed to become a NRC human factors expert?

Answer: Currently the NRC does not have an official qualification program for human-factors experts. Hiring practices ensure that, collectively, the staff has a distribution of advanced degrees in psychology (engineering, industrial, organizational, and cognitive), sociology, and industrial engineering areas, as well as field experience at operating reactors. These staff members maintain their skills and develop additional expertise by participating in continuing training which typically includes training given at the University of Michigan's Center for Occupational Health and Safety Engineering, professional workshops and seminars, in-house knowledge transfer activities, and participation in working groups and on committees addressing human factor subjects.

Question No. 92

The NRC program on human performance is described in the report. This lists a number of program elements, and the governing documents and processes, but does not indicate that internationally recognised standards and techniques are applied to regulatory activities. Please provides details of how internationally recognised human factors standards and techniques are applied to regulatory decision making.

Answer: NRC's regulatory decisionmaking for licensing reviews that concern human factors and human performance is largely guided by Chapter 18 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plant: LWR [Light-Water Reactor] Edition." For more specific guidance, this document refers to NUREG-0711, "Human Factors Engineering Program Review Model," and NUREG-0700, "Human-System Interface Design Review Guidelines." NUREG-0711 and NUREG-0700 were developed and are maintained through a formal process that involves review and careful consideration of the applicable technical-basis information. Technical bases for these guidelines are existing standards and guidelines, human-factors engineering handbooks and texts, basic literature, industry experience, and original research. As a result, these guidance documents integrate guidelines from consensus standards, including international standards; address gaps or deficiencies in available standards by drawing on other technical basis sources; and present the guidelines and criteria in a form that can be readily applied to the NRC staff reviewer's tasks. The effect is that the staff applies the guidance of many international standards to their regulatory decisionmaking, though perhaps indirectly, through their use of these review guidelines. It should also be noted that NRC policy is to give consideration to the application of existing standards, rather than develop new guidance, where the use of existing standards is deemed appropriate for NRC. Examples of consensus standards which have been endorsed by the NRC and are applicable to NRC decisionmaking include American National Standards Institute (ANSI)/American Nuclear Society (ANS) 3.2, “Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants,” and ANSI/ANS 3.5, “Nuclear Power Plant Simulators for Use in Operator Training.”

Question No. 207

What significant changes of safety review concerns were brought by the new revision of NUREG-0711, which issued on November 2012, compared with revision 2? Is there any experience obtained from the application of NUREG-0711 revision 3?

Answer: Revision 3 of NUREG-0711 does not substantially change the NRC’s human-factors review activities from those conducted under Revision 2. However, Revision 3 does incorporate lessons learned from using NUREG-0711 in several design certification reviews of new plants and includes new guidance published since the last revision. These changes are summarized in the executive summary of the document and include updated guidance for function-requirements analysis and function allocation; new guidance for the detailed design and integration of the main control room, technical support center, emergency operations facility, and local control stations; and simplification and consolidation of guidance applicable to human-factors verification and validation. Given the limited time that Revision 3 has been available, the staff has not yet accumulated significant experience in its application. However, as part of its normal process for maintaining and improving its guidance, the staff documents its application experience and seeks opportunities to gather lessons learned from external users of the guidance.

Question No. 208

12.3.5 Human Factors Information System Governing reports and Procedure.

- How does human factors information System work?
- What does NRC code, sort and retrieve the information related with Human performance?

Answer: NRC inspection reports, licensed-operator examination reports, and Licensee Event Reports are reviewed for information concerning human factors and human performance. The content is coded for source document and reactor unit, department and level of personnel involved, type of work being performed, and as pertaining to one or more of the following Categories: Training, Procedures and Reference Documents, Fitness for Duty, Oversight, Problem Identification and Resolution, Communication, Human/System Interface and Environment, and Work Planning and Practices. Each HFIS entry is also assigned an “Area” and a “Detail” code which further define the identified human-factors or human-performance issue. The Areas and Details codes can be found at http://www.nrc.gov/reactors/operating/ops-experience/human-factors/coding-scheme.html. HFIS information is maintained in a searchable and sortable database from which custom reports can be generated. Standard reports for 1998 through 2008 are publicly available at http://www.nrc.gov/reading-rm/doc-collections/human-factors/index.html.

Question No. 209

In section12 it is said “Human Factors Information System Experience. NRC program offices use the data to gain insights about human performance, to monitor the frequency of human performance issues, and to inform several types of reports, such as internal operating experience reports”. How does NRC conduct this task? What kinds of performance indicators are used?

Answer: The NRC staff uses the HFIS to produce reports summarizing the frequency of items in the database using various sorting and selection criteria and documents the analyses in an annual Technical Review Group report. The Technical Review Group establishes categories which are developed using the Reactor Oversight Program's cross-cutting areas and existing coding structures which exist in HFIS as a basis. The Technical Review Group uses the quantitative information to identify potential trends in human-performance-related events. The quantitative analysis guides the direction of the qualitative analysis. The Technical Review Group then performs in-depth qualitative reviews of subsets of the data to attempt to answer the question of why the identified trends in human-performance events occurred. The analysis focuses on the most-reported issues within the categories.
Question No. 233
According to the description of Section 12.3, the NRC performs several activities to address human performance. Please explain how many staffs with human factors expertise are working in the utility and the regulatory body, and what kinds of duties they perform in their organization.
Answer: The NRC does not require its power-reactor licensees to maintain human-factors expertise on staff and therefore does not collect information concerning site staffing of human-factors experts. NRC staff members with responsibility for human and organizational factors are principally in technical branches of NRR, RES, the Office of Enforcement (OE), and NRO. Collectively, the NRC maintains a staff of approximately 20 individuals whose principal responsibilities are to apply expertise in human and organizational factors to matters of the NRC. These staff members typically possess an advanced degree in one or more of the following areas: psychology (engineering, cognitive, and organizational), sociology, and industrial engineering. The duties that these individuals perform include, but are not limited to: review of licensee requests to amend an aspect of their licensee that might affect human performance (e.g., a proposed change to the control-room design); review of submittals for certification of new reactor designs or applications for a combined operating license; inspection of licensee programs and facilities following declines in performance below thresholds specified by the NRC's ROP (e.g., to assess licensee safety culture); inspection of licensee facilities and programs following certain operating events during which human performance might have contributed to the event or complicated the response (e.g., to assess licensee causal analysis of the event); conduct or management of research to develop the technical bases and guidelines necessary to support the NRC staff in performing the agency's regulatory functions; participation in multi-disciplinary teams to develop new NRC requirements; and advice to NRC senior management or matters of policy pertaining to human and organizational factors. In addition, the NRC also employs individuals with expertise in human-reliability analysis whose principal activities include assessing the contribution of human performance in risk assessments (e.g., application of the NRC's Significance-Determination Process) and conducting or managing research to support these activities. These individuals are also distributed among the technical branches of NRC's major program offices.

ARTICLE 13. QUALITY ASSURANCE

Each Contracting Party shall take the appropriate steps to ensure that quality assurance programmes are established and implemented with a view to providing confidence that specified requirements for all activities important to nuclear safety are satisfied throughout the life of a nuclear installation.

This section describes quality-assurance requirements and guidance for design and construction, operational activities, and staff licensing reviews. It also describes quality-assurance programs, regulatory guidance, and lessons learned from Fukushima.

Question No. 93
After Fukushima accident, were there changes for quality assurance regulations and procedures, measures of regulatory body for analysis of SAR for NPPs on the Pacific ocean coast (influence of tsunami) and NPPs near Earth' faults (influence of earthquake)?
Answer: After the Fukushima accident there were no changes to Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” to 10 CFR Part 50. The NRC is taking regulatory action in regard to flooding and earthquakes. For the latest regulatory-related information on seismic reevaluations, please visit http://www.nrc.gov/reactors/operating/ops-experience/japan-dashboard/seismic-reevaluations.html. For the latest regulatory-related information on flooding reevaluations, please visit http://www.nrc.gov/reactors/operating/ops-experience/japan-dashboard/flooding.html For the latest regulatory-related information on seismic and flooding walkdowns, please visit http://www.nrc.gov/reactors/operating/ops-experience/japan-dashboard/walkdowns.html For the latest information on emergency communications systems and staffing levels, please visit http://www.nrc.gov/reactors/operating/ops-experience/japan-dashboard/emerg-preparedness.html.
Question No. 94
It is mentioned in paragraph 13.5.1 that : " Appendix-B of 10 CFR Part 50 requires licensees that procure material, equipment or services from contractors or subcontractors to perform audits to ensure that suppliers implement an effective quality assurances program, consistent with the requirements of Appendix-B and the licensee’s technical requirement". Accordingly it is understood from Section 13.5.1 that the audit of suppliers' quality assurance program is conducted by the licensee. Please clarify how USNRC ensures that supplier maintains the acceptable quality level independently.
Answer: The NRC independently inspects vendors and suppliers under Appendix B to 10 CFR 50 on a routine and reactive basis according to the Vendor Inspection Program Plan that is available at http://pbadupws.nrc.gov/docs/ML1323/ML13239A500.pdf. In addition, the NRC routinely observes how licensees conduct their audits to assess how effective the industry is.

Question No. 210
Would you please provide detailed information on the other regulatory guidance which discusses quality assurance program controls that are appropriate for some types of nonsafety-related equipment?
Answer: Details can be found in SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," available at http://pbadupws.nrc.gov/docs/ML0037/ML003708068.pdf, and SECY-95-132, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," available at http://pbadupws.nrc.gov/docs/ML0037/ML003708005.pdf.
Question No. 211
Would you please clarify that which criterion/standard is required to be complied with to perform the "augmented quality control" to the nonsafety-related, yet important-to-safety equipment?
Answer: Details can be found in SECY-94-084, available at http://pbadupws.nrc.gov/docs/ML0037/ML003708068.pdf and SECY-95-132, and SECY-95-132, available at http://pbadupws.nrc.gov/docs/ML0037/ML003708005.pdf.
Question No. 219
The report states "Appendix B to 10 CFR Part 50 requires licensees to verify the effectiveness of their quality assurance program by performing internal audits of their programs. These audits are performed in accordance with the licensee's procedures by appropriately trained and qualified personnel who do not have direct responsibility for performing the activities being audited. The results of these audits are documented and given to management for review and corrective action". Can you elaborate on how the NRC conducts oversight of the results of internal audits?
Answer: The NRC stations resident inspectors at every operating nuclear facility whose daily jobs include reviewing the licensee's corrective-action log. Through sampling of the log, the NRC resident inspectors can evaluate how issues progress through the licensees' corrective-action program. Additionally, the NRC routinely sends out team inspections to evaluate how licensees are identifying and resolving problems. These teams use IP 71152, "Problem Identification and Resolution," which guides the examination of licensee internal audits (among other items which inspectors review). When the requirements of Appendix B to 10 CFR 50 are passed down to vendors or suppliers, the NRC routinely sends out inspection teams that use IP 43002, "Routine Inspections of Nuclear Vendors," which includes criteria that inspectors use for verifying that the requirements are met.
Question No. 234
According to Clause 13.2.3 in National Report, the NRC concluded that supplemental quality requirements would be needed when implementing Standard 9001 within the existing regulatory framework. 1. Please describe in detail what the supplemental quality requirements are. 2. Is it required to dedicate the items which are manufactured by vendors applying Standard 9001?
Answer: The NRC describes this in detail in SECY-03-0117, "Approaches for Adopting More Widely Accepted International Quality Standards," available at http://www.nrc.gov/reading-rm/doc-collections/commission/secys/2003/secy2003-0117/2003-0117scy.pdf#pagemode=bookmarks. As described in the paper, licensees could procure commercial-grade items from International Organization for Standardization (ISO) 9001 suppliers and dedicate them.

ARTICLE 14. ASSESSMENT AND VERIFICATION OF SAFETY

Each Contracting Party shall take the appropriate steps to ensure that:

(i) **comprehensive and systematic safety assessments are carried out before the construction and commissioning of a nuclear installation and throughout its life. Such assessments shall be well documented, subsequently updated in the light of operating experience and significant new safety information, and reviewed under the authority of the regulatory body**

(ii) **verification by analysis, surveillance, testing, and inspection is carried out to ensure that the physical state and the operation of nuclear installations continue to be in assurance with its design, applicable national safety requirements, and operational limits and conditions**

This section explains the governing documents and process for ensuring that systematic safety assessments are carried out during the life of the nuclear installation, including for power uprates and the PEO. It focuses on assessments performed to maintain the licensing basis of a nuclear installation. Finally, this section explains verification of the physical state and operation of the nuclear installation (by analysis, surveillance, testing, and inspection) and discusses lessons learned from Fukushima.

Other articles in this report (e.g., Articles 6, 10, 13, 18, and 19) also discuss activities to achieve safety at nuclear installations.

Question No. 95
In the report is mentioned that if the Atomic Safety and Licensing Board determines that a hearing is required, a separate legal process takes place, and NRC staff provide technical information, if needed. The safety evaluation and any hearing rulings form the basis for the NRC's final decision on the uprate request. However, the staff can authorize an uprate before the hearing is completed. The NRC issues a press release for any approved uprate. Would you provide more detail information in respect to the topics being discussed with the public and what is the procedure and outcomes if the result from a hearing is negative?
Answer: The NRC supports interested members of the public becoming informed of power-uprate applications. This is accomplished by making the application publicly available, posting power-uprate information to the NRC's public Web site, conducting public meetings, and publishing a notice of the application in the *Federal Register,* which provides an opportunity for the public to comment and request a hearing. These activities follow the same procedures as other license-amendment applications. If the application is approved, the NRC will issue a press release announcing the power uprate's approval. If the NRC receives public comments on an application within 30 days of publication in the *Federal Register*, the staff will consider the comments in making any final determination. If a member of the public requests a hearing within 60 days of publication in the *Federal Register*, the NRC will follow the hearing process. The member of the public must request a hearing and provide a contention of how the approval could affect public health and safety. The hearing process is discussed on the NRC public Web site (http://www.nrc.gov/about-nrc/regulatory/adjudicatory/hearing-pro.html) and the NRC's hearing processes are codified in 10 CFR Part 2, "Agency Rules of Practice and Procedure."

If a hearing is requested and the NRC staff has not made a final determination on the issue of "no significant hazards consideration," the NRC staff will make a final determination on the issue of no significant hazards consideration. The final determination will help decide when the hearing is held. If the final determination is that the amendment request involves no significant hazards consideration, the NRC may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment. If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

The potential negative outcomes of a public hearing process are speculative. Most hearings generally relate to safety and legal-process concerns. The staff would be required to address any negative outcomes of a hearing decision.

Question No. 96

The report discusses how regulatory approach in the US [fulfills] the goals of periodic safety reviews widely applied in the international community. One of the main goals and the basic principle of the periodic safety reviews is that licensee conducts the assessment and sends the results to the regulator for review and assessments. According to the text, periodic safety reviews are not conducted as required by the IAEA safety standards and guides (e.g. comprehensively and typically every ten years), instead it is said that the same result is reached with the other tools applied in the US regulatory approach and licensee initiatives. Based on the text it seems that it is mainly the NRC that conducts the assessments (except in license renewals and backfittings) and not the licensee. Could you clarify how licensee's role and responsibility for safety realises as expected in the PSRs in the US approach?

Answer: The NRC does not require licensees to perform periodic safety reviews (PSRs) at predefined intervals. Instead, the NRC has established processes to ensure that licensees perform continuous review and maintenance of safety of their facilities and their licensing bases. The licensing basis for nuclear power plants is established on issuance of the license and evolves throughout the term of the operating license because of the continuing regulatory activities of the NRC, as well as the activities of the licensee. As such, licensees are required to perform actions such as the following to continually maintain their licensing bases:

- Implement quality-assurance program requirements that control the procurement of materials and services and implementation of changes to facilities, processes, and procedures (Appendix B to 10 CFR Part 50). These requirements apply through the term of the license and the extended term (i.e., the term of the renewed license).
- Implement the Maintenance Rule (10 CFR 50.65) for active components. This controls the maintenance and oversight of active components through the term of the license and the extended term.
- Review plant changes in accordance with 10 CFR 50.59.
- For each NPP, implement performance- and condition-monitoring activities such as inservice testing, inservice inspection, technical specification surveillance tests, and post-maintenance operability.
- Develop and submit license amendments in accordance with 10 CFR 50.90.
- Develop and submit license-renewal applications in accordance with 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants." License-renewal applicants are required to complete an integrated plant assessment and evaluate time-limited aging analyses (TLAAs). An integrated plant assessment identifies and lists structures and components subject to an aging-management review (AMR). These include passive structures and components that perform their intended

function without moving parts or without a change in configuration or properties. Examples of these components include the reactor vessel, steam generators, piping, component supports, and seismic Category I structures.

Question No. 97

The Commission approved a modified version of option 3 to conduct a full-scope comprehensive site Level 3 PRA for an operating plant.
The plan to perform a full-scope Level-3 PRA is a challenging decision. Please provide a deeper insight into this future project of generic importance.

Answer: As stated in the "United States of America Sixth National Report for the Convention of Nuclear Safety," September 2013, the last NRC-sponsored Level 3 PRA study (NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1150/) was performed more than 20 years ago. By attempting to incorporate the many technical advances that have occurred in both PRA technology and plant safety and operational performance since completion of NUREG-1150, the full-scope site Level 3 PRA project is intended to re-baseline our understanding of nuclear power-plant risk and yield new and improved risk insights. Also, because the scope of the Level 3 PRA project includes all significant radiological sources on the site, one particular area of new insights is expected to be associated with integrated site risk.

As stated in SECY-12-0123, "Update on Staff Plans to Apply the Full-Scope Site Level 3 PRA Project Results to the NRC's Regulatory Framework" (http://pbadupws.nrc.gov/docs/ML1220/ML12202B171.pdf or http://www.nrc.gov/reading-rm/doc-collections/commission/secys/2012/2012-0123scy.pdf), potential uses of the Level 3 PRA project include:

- enhancing the technical basis for the use of risk information (e.g., by obtaining updated and enhanced understanding of plant risk and informing agency activities regarding emergency preparedness)
- improving the PRA state of practice (e.g., by demonstrating new methods for site risk assessments and supporting development of PRA screening processes)
- identifying safety and regulatory improvements (e.g., identifying potential safety improvements that might either be voluntarily implemented by licensees or lead to regulatory changes), and
- supporting knowledge management (e.g., by developing in-house PRA technical capabilities and enhancing PRA documentation practices).

It should be noted that while the Level 3 PRA project will attempt to incorporate technical advances that have occurred in roughly the last 20 years, the project is intended to be a state-of-practice study. By "state of practice," we mean that the tools, methods, and data that will be used are those that are routinely used by the NRC and licensees or have broad acceptance in the PRA technical community. As such, the Level 3 PRA project will not attempt to push the envelope of the PRA "state of the art," except in those few areas (e.g., integrated site risk) where it is deemed necessary to meet the objectives of the project.

Question No. 98

It is reported that, as of December 2012, 73 reactors (including Kewaunee) have received renewed licenses. Fifteen out of these 73 reactors have completed 40 years of operation and are operating in the extended period. Ten more reactors will also enter the period of extended operation in 2013.

Can USA indicate for any changes made in the Reactor oversight process to address any factors or parameters for these reactors which are operating in the extended period?
Answer: U.S. plants are required to adhere to all applicable rules and regulations (including license conditions and technical specification requirements) during its authorized operational life, including both the initial 40-year operating period and the PEO. The ROP continues to be applicable during the PEO. The staff has begun its review process to enhance the ROP, among other areas, to more explicitly identify inspection areas related to aging management. The enhancements are anticipated to be rolled out beginning in 2014.
Question No. 99
The process of LTO application is clearly shown in 10CFR54 & 52 in detail. With regards to the previous question, were/are there any change in the process of LTO review reflecting Fukushima accident of March 2011?
Answer: Events such as the Fukushima accident present concerns that need to be addressed in the present licensing term for all affected plants and not deferred until the time of license renewal. The NRC staff is continuing to evaluate the lessons learned from the Fukushima accident as part of the ongoing effort led by the JLD. In ways consistent with the license-renewal regulation (e.g., in 10 CFR Part 54) and the 1995 Statement of Considerations for the Final (License Renewal) Rule (as published in the *Federal Register* at 60 FR 22464), the NRC continues to rely on existing regulatory processes, such as the ongoing review efforts of the JLD, to ensure safe plant operation, while focusing license renewal only on matters uniquely relevant to the PEO. Currently, no changes to the license-renewal process in the United States have occurred as a result of the Fukushima lessons learned, because the staff anticipates that any resulting new requirements from that effort would be implemented under the existing license rather than through a future license-renewal review.
Question No. 100
Comparing the safety assessments in the US and Europe. From the NRC CNS report one gets the impression that the safety assessments in the US are largely carried out by the regulator compared to Europe, where licensees do a self assessment which is subject to review by the regulatory body. Is this a correct conclusion? What is your opinion on the different approaches?
Answer: Licensees are required to assess plant changes, plant configurations, and to provide updated information to the NRC. In addition, licensee participation is critical in the NRC ROP. The ROP is a continuous process that combines licensee monitoring and reporting of performance indicators with NRC inspection findings to get an overall assessment. NPPs' performances are assessed annually and semiannually, after which the level of NRC oversight is appropriately adjusted. The NRC performs independent assessments of plant safety as part of the ROP and its other oversight functions. In addition, the NRC performs safety assessments to justify the imposition of new regulatory requirements. The ROP contains triggers for regulatory response to emerging safety and security concerns. To get a better understanding of the ROP, please visit http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/#processdescr. A detailed ROP description and individual nuclear power plant assessment reports are available at http://www.nrc.gov/reactors/operating/oversight/rop-description.html.

The periodic safety-assessment process used in Europe to ensure the safe operation of nuclear plants is similar to the ROP in that it involves nuclear power-plant operator participation and regulatory oversight. Both processes have ensured the safe operation of nuclear power plants in their respective domains

Question No. 101

1. How does the integrated and comprehensive approach of the PSR compare with the current US approach?
2. The explicit goal of the PSRs in most European countries, is to identify potential safety improvements. Is the same true for the whole system of safety assessments taking place in the US?
3. According to the NRC its approach is as acceptable and adequate as the PSR approach. Has there been made a systematic and comprehensive comparison of the current approach and IAEA safety guide on PSR?

Answer: The NRC believes that the objectives of the periodic safety review process are well served by the totality of the current NRC regulatory process (e.g., the implementation of the ROP, which is discussed in Section 6.3.2 of the National Report, and by the review of operating experience, which is discussed in Section 6.3.5 of the National Report).

Identification of potential safety improvements is achieved through various means under the current NRC regulatory process. One way to achieve this goal is through the ROP. As indicated in Section 6.3.5 of the National Report, the performance-indicator program under the ROP offer insights into ensuring plant safety, including safety improvements.

As a part of the preparation for the IAEA IRRS Mission to the United States in 2010, the NRC compared the objective of various safety factors to the corresponding U.S. activities that accomplish these objectives. The IRRS team concluded that the NRC uses an alternative approach to meet the periodic safety-review safety factors, but suggested that we should incorporate lessons learned from periodic safety reviews performed in other countries as an input to the NRC's assessment processes (refer to Suggestion S9 in IRRS Mission report IAEA-NS-IRRS-2010/02). As stated in SECY-11-0084, the staff is conducting a periodic safety-review pilot study to review sample periodic safety-review assessment reports from other regulators. The pilot study attempts to assess various topics (e.g., aging management and safety analysis) in the periodic safety-review assessment reports that could add value to the NRC's oversight and regulatory processes.

Question No. 102

The Article describes the uprating procedure of the reactors and outcomes of implemented uprates.
How was safety justified at the reactor uprating?
Did the uprating affect the risk level of reactors (core damage frequency)?

Answer: The NRC issued Review Standard (RS-)001, "Review Standard for Extended Power Uprates," for the review and evaluation of extended power-uprate applications. RS-001 provides general guidance and specific guidance by using other NRC guidance documents as references. Because RS-001 is publicly available (ADAMS Accession No. ML033640024), licensees are aware of the guidance documents the staff uses. These guidance documents provide acceptance criteria for the areas of review.

There are many technical-review areas including: materials, chemical, mechanical, civil, and electrical engineering; reactor, plant, containment, habitability, filtration, ventilation, and instrumentation and control systems; human performance, health physics, radiological consequences, risk evaluation, and environmental assessment; and the power ascension and testing plan. The staff has used RS-001 to review applications for power uprates up to 20 percent.

Small power uprates are not expected to have an appreciable impact on risk. Extended power uprates (i.e., uprates greater than about 7 percent) can have a slight impact on the calculation of core-damage frequency. The main impact observed from extended power uprates is a slight reduction in the timing associated with performing operator actions. However, because of conservatisms in many PRAs related to timing of events (and thus conservative estimates of human-error probabilities), the impact from extended power uprates is either already bounded by the current PRA results or only slightly increased.

Question No. 103

14.1.5 The United States and Periodic Safety Reviews
Could you please explain some recent examples of NRC's more risk informed approach that have helped to ensure that resources are optimally focused on those issues more important to safety?

Answer: There are several recent examples of the NRC's use of a more risk-informed approach helping to ensure that resources are optimally focused on those issues more important to safety. Below is a summary of the NRC's use of a risk-informed approach:

- following the accident at Fukushima,
- as part of the ROP,
- in association with licensing activities under the National Fire Protection Association (NFPA) 805 standard, and
- in the exploration of the idea of enhancing safety by applying PRA to determine the risk significance of current and emerging reactor issues.

Following the accident at Fukushima in March 2011, the NRC required licensees to reassess their seismic hazard using the guidance applicable to new reactors being licensed under 10 CFR Part 52. The requirement to reassess seismic hazards includes a risk-informed screening process. If a licensee's performance-based ground-motion response spectrum is bounded by the plant's seismic design spectrum, no further action is necessary. In cases in which the ground-motion response spectrum exceeds the plant's design basis, a seismic risk assessment must be performed. Risk-informed screening criteria are employed to determine whether a licensee may use a seismic margins analysis or must perform a seismic PRA. The NRC may take regulatory action, including possible imposition of backfits, based on the results of these risk assessments.

The NRC's risk-informed approach includes not only risk insights, but also consideration of defense in depth and safety margins. Also, as a result of Fukushima, the NRC issued two orders necessary to provide reasonable assurance of adequate protection of public health and safety. The Commission noted that the events at Fukushima further highlighted the possibility that extreme natural phenomena could challenge the prevention, mitigation, and emergency-preparedness defense-in-depth layers. To address the uncertainties associated with beyond-design-basis external events, the NRC required additional defense-in-depth measures at licensed nuclear power reactors. Specifically, the NRC required licensees or construction permit holders to develop, implement, and maintain guidance and strategies to maintain or restore capabilities for core cooling, containment cooling, and spent fuel pool cooling following a beyond-design-basis external event. The second order requires that all operating BWR facilities with Mark I and Mark II containments have a reliable hardened venting capability for events that can lead to core damage. These new requirements provide greater mitigation capability in ways consistent with the overall defense-in-depth philosophy, and therefore also provide greater assurance that the challenges posed by severe external events to power reactors do not pose an undue risk to public health and safety.

In calendar year 2013, the NRR and regional staff and management performed an in-depth review of the reactor oversight program with the goal to address the following questions:

- Is the ROP helping us achieve what we need to achieve as a regulator?
- Is the ROP adequate for the current environment?
- What is the nexus between the ROP and industry safety performance?
- What is working? What is not? What should be improved?

This review was titled “Reactor Oversight Process Enhancement Project.” The staff is finalizing their report on the results of the Baseline Inspection Program portion of the ROP Enhancement Project during 2014. The goals established for this portion of the project included enhancing the baseline inspection program to incorporate the inspection areas for the current environment, eliminating inspection areas that are redundant or no longer necessary, maximizing efficient and effective use of our resources, and incorporating flexibility where appropriate. Risk information was used during the Enhancement Project to inform decisions.

In July 2004, the Commission added 10 CFR 50.48(c) to permit licensees to voluntarily adopt a risk-informed, performance-based fire-protection program. About one-half of the U.S. nuclear power plants informed the NRC of their intent to transition to NFPA 805. NFPA 805 is part of an NRC effort to incorporate risk information in the agency's regulations and enhance safety. The risk-informed performance-based approach considers risk insights as well as other factors to better focus attention and resources on design and operational issues according to their importance to safety. This approach relies on a required outcome rather than requiring a specific process or technique to achieve that outcome. It allows licensees to focus their fire-protection activities on the areas of greatest risk.

Also, the Commission has directed the staff to further explore the idea of enhancing safety by applying PRA to determine the risk significance of current and emerging reactor issues in an integrated manner and on a plant-specific basis (refer to SRM-COMGEA-12-0001/ COMWDM-12-0002, “Proposed Initiative to Improve Nuclear Safety and Regulatory Efficiency”). This would include development of approaches for allowing licensees to propose to the NRC a prioritization of the implementation of regulatory actions as an integrated set and in a way that reflects their risk significance on a plant-specific basis.

Question No. 104

As a Licensee can apply for a life extension immediately after rec[e]iving a grant for a 20 year extension, this process seem to border the ongoing license fulfillment obligations and assessments. Are there any distinct differentiations and/or rationales for a distinction between the processes?

Presumably, if the Reactor Oversight Process identifies unsurmountable shortcomings of a plant, or organisational, design this will a basis for a license revocation. There may also be a relation to regulatory required periodic safety reviews. How do these reviews interrelate and what are the distinct differences between them?

Answer: While it is true that a licensee could submit another 20-year life-extension request after it was granted a 20-year renewed license, the regulation in 10 CFR 54.17, “Filing of Application,” clearly states that no license-renewal application can be submitted earlier than 20 years before the expiration of the operating license in effect, which means that, at a minimum, the plant will have at least 20 years of operational and regulatory experience when it files its first license-renewal application. The Commission believes that 20 years should provide a licensee with substantial amounts of information and would disclose any plant-specific concerns regarding age-related degradation.

A license could be revoked if some major impediment(s) to safety were to be identified by the agency. As stated in Section 6.3.2 of the National Report, the NRC continuously oversees the nuclear power plants to ensure that plants are operated safely and in accordance with the agency’s rules and regulations. The reviews and inspections conducted under the ROP are not related to periodic safety reviews because they are not required by the U.S. regulations.

Question No. 105

What is the critical limiting factor(s) identified by NRC that would limit operational life? With aging, unceiritainty [sic] recapture, stre[t]ch power uprates and extended uprates it seems margins are eroding. How is it ensured that commensurate measures are taken to ensure this is compensated by safety enhancing measures?

Answer:

- “What is the critical limiting factor(s) identified by NRC that would limit operational life?”

  For power uprates, the limiting factors are design-specific. Generally, factors that limit extended power uprates are safety-analysis results which approach regulatory limits, structural design limits, and the design limits of the plant operating equipment (e.g., feed pumps, recirculation pumps, and transformers). The limiting components are ultimately determined by the licensee’s economic plans to install new equipment for extended power uprates. For stretch and uncertainty-recapture uprates, these are mainly justified through revised safety analysis and through the use of improved methods of evaluation.

  In terms of aging, a number of SSCs could be considered critical limiting factors when determining whether the plant can be operated safely in the PEO. Here is a representative list of the SSCs that could limit the life of the plant during the PEO:

  • Reactor pressure vessel embrittlement
  • Primary systems metals, welds and piping
  • Buried piping
  • Concrete integrity and aging
  • Cable aging

  For any of the SSCs listed here, several factors (e.g., design, material selection, fabrication, and the practices of operations and maintenance) could influence the longevity of the SSCs. For example, if proper precautions were taken during the fabrication of the critical SSC (e.g., reactor pressure vessel) to minimize stresses and other flaws, the vessel is expected to warrant a longer operational life. Similarly, if the plant operator adheres to strict primary water-chemistry guidelines, it’s less likely that the system will experience shortened life because of corrosion related to water-chemistry issues.

When an applicant submits a license-renewal application to the agency, it must demonstrate that effects of aging will be managed in such a way that the intended functions of “passive” and “long-lived” SSCs (e.g., the reactor vessel and steam generators), as identified in 10 CFR 54.4, “Scope,” will be maintained during the PEO. The NRC has developed guidance documents (e.g., the GALL Report) to guide staff’s review of the license-renewal application. The GALL Report contains numerous AMPs to age-manage those passive SSCs.

- “With aging, unceirtainty [sic] recapture, stre[t]ch power uprates and extended uprates it seems margins are eroding.”

  Regarding power uprates, design margins have been reduced in respect to regulatory limits in many cases. However, provided that the facility operates within the regulatory limits, the design meets the Commission’s standards for adequate protection of public health and safety. For both power-uprate and license-renewal programs, the agency has programs in place to monitor and respond to industry performance trends that could adversely affect safety.

- “How is it ensured that commensurate measures are taken to ensure this is compensated by safety enhancing measures?”

  Provided that the plant meets its existing design basis and addresses the regulatory standards for uprate and license renewal, no safety enhancements are mandated. For uprates and license renewal, limiting plant components are sometimes replaced with newer equipment that has higher capacity and reliability. This newer equipment often improves overall plant safety.

  For both power uprate and license renewal, operating experience has been incorporated in the NRC’s technical review. If warranted, the staff can condition approvals for both power uprate and license renewal to require licensees to take additional steps to ensure safety. Additionally, through the NRC inspection program, equipment performance is monitored through performance indicators. Should plant equipment reliability decrease, additional NRC inspection will occur to ensure that the nuclear power plant’s equipment performance is adequately maintained.

Question No. 106

Were SAMG revised after Fukushima Daiichi accident? What is the accepted scope of SAMG? How is the qualification issue being resolved for the equipment involved in severe accident management?

Answer: For the emergency onsite response capabilities rulemaking, initiated as a result of NTTF Recommendation 8 (i.e., strengthening and integration of emergency operating procedures, SAMGs, and extensive damage-mitigation guidelines), the NRC issued a draft regulatory basis for public comment on January 8, 2013. The staff is currently considering both internal and external feedback and modifying the document. The final rule, when complete, is expected to establish standards that ensure that plants can smoothly transition between various emergency procedures, keeping overall strategies coherent and comprehensive. The final rule is scheduled for completion in March 2016.

This is discussed in Enclosure 1, “Update on Tier 1 Activities,” to SECY-13-0095, http://www.nrc.gov/reading-rm/doc-collections/commission/secys/2013/2013-0095scy.pdf

It is desirable for diverse mitigation equipment to be commonly available (e.g., commercial equipment) so that parts and replacements can be readily obtained.

Equipment qualification is discussed in the NRC-endorsed NEI 12-06, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide," http://pbadupws.nrc.gov/docs/ML1224/ML12242A378.pdf.

Question No. 107

Indicated that after a licensee submits an power uprate application, the NRC issues a Federal Register notice to alert the public that the agency is considering the application. The public has 30 days to comment on the licensee's request and 60 days to request a hearing where the application could be contested.
Will be a similar procedure for public hearing on measures to improve NPP's safety?
Is there formal list of NPP's safety related technical and organizational changes which are subject for public notification and hearing?

Answer: All applications for changes to a nuclear power plant are processed in a similar manner. Facility changes, regardless of whether they increase or decrease safety margin, are reviewed to determine whether a license amendment is required before proceeding. The changes would be reviewed against the facility-change process using 10 CFR 50.59.

Plant modifications that require technical specifications changes always require a license amendment. More minor changes would be evaluated using the 10 CFR 50.59 process. The licensee's 50.59 evaluation would review the impact of the change with regards to how it alters the existing licensing and design basis of the nuclear power plants. As a result, some safety enhancements will require a license amendment and others are within licensee control to implement on their own. For most cases, the licensee can proceed under 50.59 to implement enhancements provided that they do not dramatically alter the existing licensed design.

A license-amendment process affords both a notification in the *Federal Register* and a hearing opportunity for members of the public. There are similar provisions for submittal of the initial license request.

Question No. 108

It is noted that there is a significant trend to extend the operating life of many reactors in the US. Major areas of research are described in respect of this for subsequent license renewal.. However, obsolescence of components is not described in the report. Has obsolescence been identified as a major issue in technical areas such as instrumentation and control equipment, particularly in aging plants, and if so, what program is applied to regulate the effects of this, and to encourage proactive management of obsolescent components amongst licensees?

Answer: Managing aging of nuclear power plants implies ensuring the availability of required safety functions throughout the life cycle of the plant, including the changes that occur over time and expected wear and tear. This requires addressing both physical aging of SSCs, which results in degradation of their performance characteristics, and obsolescence of SSCs, which results in them becoming outdated with respect to the then-current knowledge, standards, and technology.

The GALL Report focuses on management of physical aging. Obsolescence of SSCs important to safety is managed proactively throughout their service life. Aspects of technological obsolescence, such as insights into individual degradation mechanisms, have already been taken into consideration in the GALL AMPs. New insights will be addressed in future updates of the AMPs.

Question No. 109
It is noted that a significant number of reactors in the US are operating, or will shortly be operating in the extended period. The report describes activities in relation to this, including work with other national regulators, and research. How is the NRC carrying out oversight of research activities to ensure targeted and effective use of resources, and to ensure that learning is propagated throughout the industry?
<u>Answer</u>: RES is a major research program office mandated by the U.S. Congress. The office plans, recommends, and implements a program of nuclear regulatory research, standards development, and resolution of generic safety issues for nuclear power plants and other facilities regulated by the NRC. The office accomplishes its regulatory research mission by conducting research both in house and with the use of contractors. One method to ensure that research activities are targeted and effective involves a process by which the regulatory offices describe the regulatory issue and proposed research activities in a user need request. Over three-fourths of its activities are driven by the direct needs of the NRC's regulatory offices. More specifically, with respect to nuclear power plants' license renewal and potential operation for longer than 60 years, the NRC staff is presently updating the original NUREG/CR-6923, "Expert Panel Report on Proactive Materials Degradation Assessment," issued in February 2007, to include longer timeframes (i.e., 80 or more years) and passive, long-lived components beyond the primary piping and core internals, such as the pressure vessel, concrete containment building, and cable insulation. This update will allow staff to (1) identify significant knowledge gaps and any new forms of degradation that might have emerged since the original proactive materials-degradation assessment report was developed; (2) capture the current knowledge base on materials-degradation mechanisms; and (3) prioritize materials-degradation research needs and directions for future efforts. The NRC staff is accomplishing this task through a collaborative effort with the U.S. DOE's light-water reactor sustainability program and expects to complete the task in 2014. This report will ensure targeted and effective use of the NRC's resources for its confirmatory work on the research conducted by industry to provide the technical basis for operation in the second extended period.
Question No. 110
Section 14.1.5 describes the process by which the NRC delivers ongoing oversight of the status of nuclear power plants using a variety of routine activities. However, routine activities have the potential to become focussed on piecemeal improvements, and to fail to identify new approaches that could improve safety. Periodic Safety Reviews provide an opportunity to stand back and review the totality of the measures providing safety, and to identify additional measures that are beyond existing programs and activities. Please describe how the NRC works to ensure the adequacy of improvement programmes to address broader safety issues
<u>Answer</u>: The U.S. regulatory approach provides a continuum of assessment and review that ensures public health and safety throughout the period of plant operation. While some of this is achieved through routine activities, certain aspects of the comprehensive regulatory process are based on a broader assessment of issues. As an example, the NRC's Generic Issues Program and Generic Communication processes are used to continually evaluate industry-wide safety-significant issues that might require a technical solution. When issues are identified, the agency issues generic communications such as Bulletins, GLs, RISs, and INs to alert licensees of the issues. Where there is a potential safety measure that might be beyond existing programs and activities, the staff engages licensees or industry for resolution and can impose new requirements if they meet the backfit

criteria. Through these deliberative processes, the NRC is able to ensure that safety issues are identified along with their appropriate resolution.

Question No. 111

It is reported in p.18 that “In the fifth report, seven units entered into their 41st year of operation, and additional 18units will have entered the period of extended operation.”

The process of LTO application is clearly shown in 10CFR54 & 52 in detail.
Concerning the process of review the LTO application of these units, are there any common points emphasized in the inspection related to long term operation?

Answer: There are several inspection stages that occur throughout the license-renewal review process and continue into the PEO that have different scopes and objectives.

The license-renewal application review consists of an in-office acceptance review, an onsite scoping and screening audit, an onsite AMP audit, and an in-office audit of TLAAs. The license-renewal application is primarily reviewed by NRR to verify that the content of the application meets the technical and regulatory requirements of the license-renewal rule.

In addition, site inspections are performed before renewal of the operating license to assess the applicant’s implementation of and compliance with 10 CFR Part 54 requirements. The application site inspection activities will be performed based on IP 71002, “License Renewal Inspection.” The NRC Regional offices are responsible for the team inspection. The IP 71002 inspection verifies the implementation of (or readiness to implement) license-renewal activities.

Once the renewed operating license is issued, but before the licensee begins operating in the PEO, inspections are conducted in accordance with IP 71003, “Post-Approval Site Inspection for License Renewal.” The post-approval site inspections will be performed by a team to verify that the license conditions added as part of the renewed operating license, regulatory commitments, selected AMPs, and TLAAs are adequately implemented and/or completed. The inspection also verifies that (1) the updated final safety analysis report (UFSAR) includes any “newly identified” SSCs that should have been within the scope of the license-renewal program and subject to an AMR or TLAA evaluation under 10 CFR 54.37(b); (2) the description of the AMPs and related activities are, or will be, contained in the UFSAR; and (3) the description of the programs is consistent with how the programs are implemented by the licensee. The inspection team will verify that the licensee submitted a license-amendment request to the NRC staff in accordance with 10 CFR 50.90 for changes to a license condition for license renewal, managed changes to the UFSAR supplement in accordance with 10 CFR 50.59, and managed changes to regulatory commitments associated with license renewal in accordance with NEI 99-04, “Guidelines for Managing NRC Commitment Changes,” as endorsed by RIS 2000-017, “Managing Regulatory Commitments Made by Power Reactor Licensees to the NRC Staff.” The Regions are responsible for the team inspection.

Once the licensee enters the PEO, the scope of the inspections are more performance-based than program-based and focus on assessing the licensee’s management of age-related degradation. Three inspection procedures have been updated thus far. More procedures will be updated to incorporate performance-based inspections of age-related degradation issues identified during the period of extended operation to ensure that the licensee verifies that the issues related to aging of equipment within the scope of license renewal are identified and properly resolved in a timely manner.

Question No. 112
The USA presents its contribution to some international symposia on Long Term Operation (LTO). Have there been any IAEA SALTO missions to some of the ageing NPPs in the USA?
<u>Answer</u>: To date, none of the U.S. plants has received an IAEA Safety Aspects of Long-Term Operation of Water-Moderated Reactors (SALTO) mission visit yet.
Question No. 113
The requirement seems to be based on probabilistic methods, how is deterministic analysis taken into account?
<u>Answer</u>: The requirement in 10 CFR 50.55a, “Codes and Standards,” as discussed in Section 14.2 of the report, is not based on probabilistic methods; rather, it is based on deterministic analyses because the Commission believes that the methodology for conducting a safety analysis needed to ensure the plant’s safety should be deterministic in nature. In addition, other testing and surveillance programs (such as the plant technical specifications and inservice testing) are typically based on deterministic criteria. However, in recent years, several initiatives have been documented in RGs to describe processes to risk-inform various testing, surveillance, and inspection programs, such as: • RG 1.174; • RG 1.175, “An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing”; • RG 1.177, “An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications”; and • RG 1.178, “An Approach for Plant-Specific Risk-Informed Decisionmaking for Inservice Inspection of Piping.” The NRC’s inspection program is risk-informed in such a way that inspection samples are selected based on risk insights. Below are some examples of baseline IPs that pertain to testing, surveillance, and analyses: • IP 71111.08, “Inservice Inspection Activities”; • IP 71111.15, “Operability Determinations and Functionality Assessments”; • IP 71111.18, “Plant Modifications”; • IP 71111.19, “Post Maintenance Testing”; • IP 71111.21, “Component Design Bases Inspection”; and • IP 71111.22, “Surveillance Testing.”
Question No. 212
“The commission has decided that no final licenses will be issued until a new Waste Confidence Decision and Rule are in effect; “ Please describe the main effect of the new Waste Confidence Decision and Rule on the application of license renewal.
<u>Answer</u>: The main practical effect is that until the new Waste Confidence Rule is finalized and codified as a regulation, the NRC cannot issue a renewed operating license for a nuclear power reactor. Once the new Waste Confidence Rule is in effect, the NRC can then issue renewed operating licenses to those applicants that have completed all regulatory requirements. The NRC continues to perform its safety and environmental reviews of license-renewal applications concurrently with the rulemaking process for the new Waste Confidence Rule. The new Waste Confidence Rule and the final Waste Confidence Environmental Impact Statement,

which provides the technical basis for the rule, are expected to be completed in November 2014.

Question No. 213

One of important target of 10 years' Periodical safety review is to evaluate the influence of safety capacity from the accumulate aging effects. However, nuclear power plants do not use PSR in the United States. So how to evaluate the accumulate aging effects for the safety capacity? What specific technology and measurement have been developed for those supervise and evaluation?

Answer: When an applicant submits a license-renewal application to the agency, it must demonstrate that effects of aging will be managed in such a way that the intended functions of “passive and long-lived” SSCs (e.g., the reactor vessel and steam generators), as identified in 10 CFR 54.4, will be maintained during the PEO. The NRC has developed guidance documents (e.g., the GALL Report) to guide staff’s review of the license-renewal application.

The GALL Report contains numerous AMPs to detect and/or manage the aging effects of the reactor systems. For instance, AMP X.M1, “Fatigue Monitoring,” in the GALL Report describes how to calculate fatigue usage factor from suitable inputs as a parameter suitable for gauging fatigue damage in components subjected to fluctuating stresses.

A second example is the various nondestructive examination techniques (e.g., visual and ultrasonic examinations) used to conduct the inservice inspections as required to meet the requirements of the ASME Codes. Table 1 in RG 1.147, “Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1” lists some of those techniques. These nondestructive examinations are commonly cited in various AMPs (e.g., XI.M32, “One-Time Inspection”) in the GALL Report to ascertain material conditions. The NRC staff believes that the cumulative aging effects can be properly detected by the appropriate application of these nondestructive examinations.

Question No. 220

The report states “The reviews include an assessment of plant design and operation against current safety standards and practices, with the objective of ensuring a high level of safety throughout the plant’s operating lifetime”. If gaps are identified between the two standards (current and during licensing), how does NRC address the gaps?

Answer: The NRC constantly evaluates operating experiences from domestic and international plants to assess the adequacy of safe plant operations. The NRC also participates in the standard-setting bodies’ activities to stay abreast of the current standards. If significant gaps between the current version of the standard and the standard under which the plant was licensed were to exist, the NRC would first evaluate the difference(s) between the two versions of the standards and the associated safety significance of any differences. Depending on the outcome of the NRC evaluation, the agency may request that the licensee demonstrate how it ensures safety by meeting the more stringent version of the standards. The agency will evaluate the licensee’s response for adequacy before rendering a decision.

Question No. 221

The report states under the heading “License Renewal Confirms Safety of Plants” that “The current licensing basis must be maintained throughout the period of extended operation“, can you clarify what is meant by “current licensing basis”? For example, the original licensing basis used for commissioning of the plant, or an updated version resulting from some issues or new regulatory requirements?

Answer: The current licensing basis is the set of NRC requirements applicable to a specific plant and the licensee's written commitments for ensuring compliance with, and operation within, applicable NRC requirements and the plant-specific design basis (including all modifications and additions to such commitments over the life of the license) that are docketed and in effect. Because the current licensing basis might evolve over the life of the license, as in the case of modifications made to plant since the initial commissioning of the plant, the licensee is required to maintain the current licensing basis throughout the entire life of the license (both the initial license period and the PEO).
Question No. 222
Under "Governing Documents and Process", please clarify what is meant by "the licensing basis of all currently operating plants provides an acceptable level of safety"? Are probabilistic safety goals included in the licensing basis of operating NPPs? If yes, does it also imply, for example, that Core Damage Frequency (CDF) should be 10-4 or 10-5 for "acceptable level" for 20 years licence renewal?
Answer: This statement comes from the first principle of license renewal. The first principle basically states that with the exception of age-related degradation unique to license renewal and possibly a few other issues related to safety only during the PEO of nuclear power plants, the regulatory process is adequate to ensure that the licensing bases of all currently operating plants provides and maintains an acceptable level of safety so that their operation will not be inimical to public health and safety or to the common defense and security. Moreover, consideration of the range of issues relevant only to extended operation led the Commission to conclude that the detrimental effects of aging are probably the only issue generally applicable to all plants. As a result, continuing this regulatory process in the future will ensure that this principle remains valid during any PEO if the regulatory process is modified to address age-related degradation that is of unique relevance to license renewal. No probabilistic safety goals are explicitly included in the licensing basis of nuclear power plants for the PEO because the Commission believes that the methodology for conducting an integrated plant assessment needed to ensure the appropriate management of SSCs' aging should be deterministic in nature. Nevertheless, the NRC recognizes that a plant-specific PRA can be used as an effective tool to provide integrated insights into the plant design, resulting in an additional relative measure of overall plant safety. In addition, the Commission also acknowledges that PRA can be an effective tool to provide added assurance that all SSCs important to license renewal have been appropriately evaluated.
Question No. 223
The report states that "The Atomic Energy Act and NRC regulations limit commercial power reactor licenses to 40 years but permit such licenses to be renewed", please clarify at which point in time does the 40-year licence period start (i.e. does it correspond to the date of licence issuance, or the date of the start of commissioning, or the start of commercial operation? These dates can differ by few years.
Answer: The 40-year license period starts when NRC issues an operating license to a plant. Typically, these are the key dates associated with any nuclear power plant in the United States: 1. The date of the issuance of the construction permit, 2. The date of the issuance of the operating license, 3. The date of the start of commercial operation, 4. The date of license renewal (if the NRC has already renewed its operating license), and 5. The date of license expiration.

Here is an example: For Susquehanna Steam Electric Station Unit 1 (Berwick, PA), the construction permit was issued on November 3, 1973. The operating license was issued on July 17, 1982; commercial operation began on June 8, 1983. The NRC renewed its license on November 24, 2009, and the license expires on July 17, 2042, or sixty years (40 years of its original license period plus an additional 20 years of its renewed period) after its original operating-license date of July 17, 1982.
Question No. 235
The description and philosophy of section 14.1 could be applied to new generation reactors. Could you describe the different aspects in licensing and regulation for the new generation reactors?
Answer: At the moment our regulations apply to all reactors. We do not have different regulations for different designs or generations. We realize that as technology changes we might need to address gaps.

ARTICLE 15. RADIATION PROTECTION

Each Contracting Party shall take the appropriate steps to ensure that, in all operational states, the radiation exposure to the workers and to the public caused by a nuclear installation shall be kept as low as reasonably achievable, and that no individual shall be exposed to radiation doses which exceed the prescribed national dose limits.

This section summarizes the authorities and principles of radiation protection, which include the regulatory framework, regulations, and radiation-protection programs for controlling radiation exposure for occupational workers and members of the public. This section also discusses lessons learned from Fukushima. Article 17 of this report discusses radiological assessments that apply to licensing and facility changes.

Question No. 114
Does the US NRC record contamination events occurring in NPPs? Is this experience feedback used to define a policy in the field of the control of radioactive cleanliness?
Answer: The NRC does not have any specific contamination limits and does not keep records on contamination events at nuclear power plants, but does limit the resulting doses to workers and the public. Licensees are required to implement radiation-protection procedures, including contamination control, that are sufficient to ensure that the requirements in 10 CFR Part 20, “Standards for Protection against Radiation,” (e.g., those for dose limits from internal contamination and release of licensed material from the site) are complied with. Under conditions specified in 10 CFR 50.72(b)(2)(xi), licensees are required to submit reports to the NRC when certain types of groundwater contamination events occur. The NRC maintains these event-notification reports. Additionally, under the conditions specified in NRC regulations at 10 CFR 50.75(g), licensees are required to keep records of information important to the safe and effective decommissioning of the facility (e.g., when significant leaks or spills to the environment occur). Further, the NRC may inspect licensee records during reviews of radiation-protection programs, and environmental programs, and follow up on events as appropriate.
Question No. 115
Could the USA specify if the effluent release optimization is based on the BAT (best available technology) principle? Could the USA give some examples of provisions liable to contribute to the reduction of worker exposure and some examples of provisions which have contributed to the reduction of liquid and gaseous releases into the environment?
Answer: The NRC implements an “as low as reasonably achievable” (ALARA) concept (e.g., 10 CFR 50.34a, 10 CFR 50.36a(a), and Section II.D of Appendix I, “Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion ‘As Low as is Reasonably Achievable’ for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents,” to 10 CFR 50) that takes into account the state of technology and the economics of improvement. Although ALARA and best-available techniques share some similarities, there are significant differences, particularly in implementation. ALARA has contributed to the reduction of liquid and gaseous releases into the environment. For example, at the time of licensing, licensees are required (by Appendix I to 10 CFR 50) to

demonstrate, by a cost-benefit calculation, that adding additional effluent processing equipment will not result in a reduction in public collective dose by 1 person-rem for every $1000 spent for populations located with an 80-km radius of the reactor site. If adding specific processing equipment (for liquid and gaseous effluent treatment) will reduce the collective public dose by more than 1 person-rem per $1000, the licensee is required to augment the processing equipment for that effluent stream to ensure that radioactive releases are ALARA. A significant impact on both occupational dose and effluent releases has been achieved at U.S. nuclear power plants by lowering the radioactive source terms resulting from improved the reactor fuel integrity performance and the implementation of better primary and secondary coolant chemistry controls. Additionally, in accordance with the NRC guidance and regulations, licensees have incorporated provisions into their Technical Specifications that require approved waste-processing systems to be used when liquid or gaseous effluents exceed a small fraction of the regulatory dose limit for members of the public (10 CFR 20.1301, "Dose Limits for Individual Members of the Public").

Worker exposure has also decreased as a result of senior plant management directly supporting the ALARA program, to include improved job planning and implementation. A good example includes improvements in shutdown chemistry control associated with refueling and maintenance outages (e.g., improved reactor lithium control, improved degassing of primary systems, hydrogen peroxide injections at PWRs, and flushing and draining practices). This has led to a reduction in both dose rates and person-hours at the job sites.

Question No. 116

Please give information on automatic radiation monitoring stations in the vicinity of NPP's?

Answer: Note: NRC staff interprets the term "automatic radiation monitoring" as being continuous, remote offsite radiation monitoring with telemetry to report data back to a central monitoring location.

- NRC (as the regulator) does not perform its own automatic radiation monitoring (or any other routine environmental monitoring).

- NRC does not require nuclear power plants to perform automatic radiation monitoring.

  Voluntarily, a few nuclear power plants have installed automatic radiation-monitoring systems. The monitoring data is typically transmitted back to plant computers through telemetry.

  – The nuclear power plants that have automatic monitoring systems include the Susquehanna, Diablo Canyon, San Onofre, Arkansas Nuclear, and Indian Point plants.
  – Some plants use the automatic monitoring data in their Emergency Plan.
  – Some plants share the monitoring data with their State and local authorities during emergency drills.

  A few U.S. State governments have automatic radiation monitoring in the 10-mile emergency planning zone with data telemetered back to State offices.

  – This includes the states of New Jersey, Illinois, Alabama, Delaware, Massachusetts, and Vermont.

- Public-Interest Groups

A few public-interest groups operate their own automatic radiation-monitoring systems. The names of the public interest groups are:

– Decommission San Onofre (http://decommission.sanonofre.com/)
– C-10 Research and Education Foundation (http://www.c-10.org/)
– Three Mile Island Alert (http://www.tmia.com/about)
– Some public-interest groups provide their radiation monitoring data on public web sites such as radiationnetwork.com.

- Further, the U.S. Environmental Protection Agency (EPA) maintains a network of radiation monitors across the United States, some of which might be in the vicinity of nuclear power plants (http://epa.gov/radnet/).

Question No. 117

a) What are the present results of the assessments of public exposure by the USA plants?
b) Can you also give some detailed information on the level of discharges of the operating plants and the limits and conditions of discharges?

Public exposure near U.S. nuclear power plants from routine operations has always been within the ALARA criteria of 10 CFR 50, Appendix I.

Each nuclear power plant prepares an annual assessment of effluent discharges and public radiation exposure. Each licensee reports the information in its Annual Radiological Effluent Release Report. These reports are publicly available on the NRC web site at: http://www.nrc.gov/reactors/operating/ops-experience/tritium/plant-info.html.

In addition, the NRC publishes an annual summary report, NUREG/CR-2907, that summarizes the effluent releases and dose assessments for the operating nuclear power plants in the United States. These annual summary reports have the ADAMS Accession Numbers listed below.

- 2009 report: ML13218A300
- 2008 report: ML103620452
- 2007 report: ML103620453

The plant's operating limits for radioactive effluent discharges are specified in the plant technical specifications. These operating limits (and reporting requirements) are established to ensure that the plant's effluent discharges will be ALARA and meet the design objectives of 10 CFR 50, Appendix I.

Question No. 118

Why is USA using radiation protection rules which are relatively old? For instance a dose limit of 50 mSv is not in line with current international standards and common practice in other states. Why is lowering the occupational dose limit and using recent models not seen as a safety improvement?

Answer: The current limit on occupational total effective dose equivalent is 50 millisieverts (mSv) (5 rem) per year. This value is the same as the single-year value recommended by the International Commission on Radiological Protection (ICRP).

Regulatory requirements in the United States, including those of the NRC, must be completed through a rulemaking process which includes notice for public comment. A rulemaking must include a regulatory analysis of benefits and impacts of the proposed action, and, in addition, a “backfit” analysis as required in 10 CFR 50.109; 10 CFR 52.39, 52.63, 52.83, 52.98, 52.145, and 52.171; Section VIII of Appendices A, B, C, and D to 10 CFR 52; 54.37; 10 CFR 70.72; 10 CFR 72.62; and 10 CFR 76.76 (for titles, see http://www.nrc.gov/reading-rm/doc-collections/cfr/). A backfit analysis requires that there be an assessment of benefits and impacts, and that the proposal is justified by those benefits, unless the requirement is considered as necessary for adequate protection. The NRC staff continues with its development of a regulatory basis for possible changes in ways consistent with Commission direction.

The NRC staff, in SECY-12-0064, “Recommendations for Policy and Technical Direction to Revise Radiation Protection Regulations and Guidance,” had proposed to the Commission that it continue to explore the implications of lowering the dose limit. During the Commission’s deliberations, the ACRS also reviewed the subject; provided a letter to the Commission, dated October 16, 2012; and concluded that rulemaking to revise the limits for occupational radiation exposure should not be undertaken. That letter stated that a decision to change dose limits should be based on demonstrated health and safety benefits and consideration of negative, unintended safety consequences. The letter stated that there is disagreement among professional organizations involved in radiation protection regarding the health and safety benefits of reducing the dose limit from 50 mSv (5 rem) to 20 mSv (2 rem) per year. In the absence of a clear and well-demonstrated benefit, the ACRS disagreed with lowering the dose limit. The ACRS also stated that compliance with lower dose limits could also have unintended negative consequences. Finally, the ACRS noted the excellent performance of the nuclear power industry in implementing the ALARA principle, which has resulted in exposures well within the national and international recommendations. Given the industry performance, the ACRS recommended that the staff develop improvements in the NRC guidance for more effective implementation of ALARA strategies and programs in those segments of the regulated community that have identified compliance problems. In summary, because exposures in the nuclear power industry have been reduced through ALARA implementation, a change in the limits would not substantially change the actual exposure of occupational workers, and could entail potentially significant costs and negative consequences.

The Commission, in SRM-SECY-12-0064 (which, as most SRMs do, shares the title of the SECY paper to which it responds), disapproved the staff’s recommendation to develop the regulatory basis to reduce the occupational total effective dose equivalent. The SRM also directed the staff to continue discussions with stakeholders on alternative approaches to deal with individual protection at or near the current dose limit. The Commission approved the staff developing the regulatory basis to adopt the updated methodology and terminology of ICRP Publication 103, “The 2007 Recommendations of the International Commission on Radiological Protection.” The NRC staff is currently preparing an Advance Notice of Proposed Rulemaking to specifically solicit further public input on the issues of updating methodology and terminology, ALARA implementation, and possible additions to ALARA requirements in response to the Commission’s direction.

Question No. 119

RASCAL: Is it planned to use new codes for calculating workers doses for local actions in extremely adverse radiological conditions?

Answer: RASCAL is not suited for calculating worker doses for local actions in extremely adverse radiological conditions. However, the Turbo FRMAC 2013 computer code, developed by Sandia National Laboratory, has the capability to establish worker-protection guidelines. It uses methods documented in the Federal Radiological Monitoring and Assessment Center Assessment Manual, which guides the response of Federal, State, local, and Tribal

governments in the United States. The NRC is scheduled to get the 2013 FRMAC update in the next month or so.
Question No. 120
Who is responsible for keeping dosimetry records for contractors' workers performing tasks in NPPs?
Answer: Under the requirements of 10 CFR 20.2106, "Records of Individual Monitoring Results," licensees are required to maintain the records of all individuals (including contract workers) for whom radiation monitoring (dosimetry) was provided under 10 CFR 20.1502, "Conditions Requiring Individual Monitoring of External and Internal Occupational Dose." Nuclear power-plant licensees are required to report this data annually to the NRC. The NRC then compiles an annual dose report in NUREG-0713, "Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities" (refer to http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0713/).
Question No. 121
Who is responsible for keeping dosimetry records for contractors' workers performing tasks in NPPs?
Answer: *[Repeated question]* Under the requirements of 10 CFR 20.2106, licensees are required to maintain the records of all individuals (including contract workers) for whom radiation monitoring (dosimetry) was provided under 10 CFR 20.1502. Nuclear power-plant licensees are required to report this data annually to the NRC. The NRC then compiles an annual dose report in NUREG-0713.
Question No. 122
The report states that "The NRC judges complience with requirement on the basis of a licensee´s capability to track and, if necessary, reduce exposures, rather than on whether exposures and doses represent an absolute minimum or whether the licensee had used all possible methodes to reduce exposure". Is this an exception from ALARA, and how is that motivated?
Answer: The NRC requires its licensees (in 10 CFR 20.1101, "Radiation Protection Programs") to include the use of procedures and engineering controls to achieve radiation doses that are ALARA in their radiation-protection programs. This requirement was established in the 1991 revision to 10 CFR 20. The statement quoted in the question is taken from the Statements of Consideration notification published in the *Federal Register* (Vol. 56, No. 98, Tuesday, May 21, 1991) that announced the issuance of the revised Part 20. The NRC included this statement to clarify that the new requirement was a requirement to establish a reasonable program with the objective of reducing radiation doses to ALARA. It was not intended that every radiation dose received was required to be an absolute minimum. The statement does not represent "an exception from ALARA"; it clarifies the intent of the new regulation. The required programs are expected to achieve results (i.e., doses that are ALARA) as judged by the overall collective dose trends.
Question No. 123
Is the ALARA program reviewed by the regulatory body?
Answer: Yes. NRC IP 71124.02, "Occupational ALARA Planning and Controls," directs the inspector to review the licensee's ALARA program for planning and implementing radiation-protection work controls.
Question No. 124
How does the NRC review compliance with exsisting dose limits if the license holder is not trequired to report them?

Answer: Licensees are required to maintain records to demonstrate compliance with regulatory requirements. The NRC may, and does, inspect these records during inspections of licensee activities.

Licensees are required to report any exceedance of dose limits in accordance with 10 CFR 20.2202, "Notification of Incidents." An immediate notification is required if the exceedance is five times the limits and a 24-hour notification is required for any lesser exceedance of a limit.

In addition, the NRC requires licensees to file a written report within 30 days of any incident involving an exceedance of an occupational or public dose limit in accordance with regulations at 10 CFR 20.2203, "Reports of Exposures, Radiation Levels, and Concentrations of Radioactive Material Exceeding the Constraints or Limits."

Question No. 125

What is the average collective, yearly, dose for a BWR and a PWR?

Answer: In 2011 (from the data compilation in NUREG-0713, Volume 33), the average collective dose per reactor for PWRs was 55 person-rem and the average collective dose per reactor for BWRs was 142 person-rem.

Question No. 126

Is the licencees required to prove the efficiency and function of measurement equipment and release liminting-systems on a regular basis?

Answer: Applicants for new reactor licenses and licensees are required to demonstrate the effectiveness of methods used to control and monitor (through liquid- and gaseous-effluent processing systems) during the licensing-review process and plant operations. The treatment systems are determined to be acceptable if they meet the dose based design criteria and ALARA provisions of Appendix I to 10 CFR 50. For operating reactors, the NRC inspects the operational programs and plant operations biennially to ensure that the licensees operate within this design basis, in addition to those effluent concentration limits in Table 2 in Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage," to 10 CFR 20. In addition, licensees must annually report the levels of effluent discharges to the environment in the previous year and demonstrate they continue to meet the criteria in Appendix I to 10 CFR 50. The regulation in 10 CFR 20.1501(c) ensures that equipment used to quantify radiation measurements and effluent concentrations (including process monitors and sampling and offline analysis) are periodically calibrated for radiation levels and radionuclide distributions and concentrations expected in effluents.

Question No. 127

How does the NRC review the data of releases and resultant doses accomplished by the licencees (by independent monitoring and dose calculations or in some other way)?

Answer: The NRC performs biennial inspections of licensees' effluent-release programs under its IP 71124.06, "Radioactive Gaseous and Liquid Effluent Treatment" (ADAMS Accession No. ML092810408). Inspectors can review the data by:

- Using the NRC's PC-DOSE computer code,
- performing manual calculations, or
- reviewing the licensee's dose calculation methods.

Additionally, the NRC staff and inspectors can compare any site's effluent data to the typical industry values and industry trends published by the NRC (e.g., NUREG/CR-2907, Volume 15, "Radioactive Effluents from Nuclear Power Plants: Annual Report 2009," ADAMS Accession No. ML13218A300; also see companion documents which summarize 2007 effluents (ML103620453) and 2008 effluents (ML103620452)). This type of comparison can identify problems that could not be identified by simply checking the mathematical equations and calculations.

Question No. 128

"Provide data on measurable levels of radiation and radioactive materials in the environment to evaluate the relationship between quantities of radioactive material released in effluents and resultant radiation doses to individuals from principal pathways of exposure"

Do you take account of the impact of tritium and C-14 when assessing the radiation dose to the critical group of the population from NPP releases?

Answer: Yes, the dose impact of tritium and C-14 are included in public dose assessments. Regulatory guidance on determining the radiation exposure of the public is given in RG 1.21, "Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste" (ADAMS Accession No. ML091170109).

In the RG, tritium releases from normal and abnormal discharges are discussed extensively. C-14 generation and accounting is discussed specifically in Section C, paragraph 1.9, "Carbon-14." The quantity of C-14 discharged can be estimated by sample measurements or by use of a normalized C-14 source term and scaling factors based on power generation. Sections 5.7 through 5.9 provide information on dose assessments that includes information about C-14 and tritium.

Each nuclear power plant provides a report to the NRC on annual basis with the amount of radioactive effluents released and an assessment of public doses from those releases. These nuclear power-plant reports are published on the NRC's Web site at http://www.nrc.gov/reactors/operating/ops-experience/tritium/plant-info.html.

Similarly, each individual plant publishes an annual environmental report showing measurements of radioactive materials in the environment. These environmental reports are also provided at the above NRC Web site. The NRC also publishes an annual report summarizing and comparing the nuclear plant releases in NUREG/CR-2907 (http://pbadupws.nrc.gov/docs/ML1321/ML13218A300.pdf).

Question No. 189

Section 15.4.2 describes some of the activities related to NRC response to reports of ground water contamination. The report states that there have been numerous occurrences of buried piping leaks; however the health and safety impacts to the public were not significant. The releases were below regulatory limits. No regulatory changes are contemplated. How has the NRC considered ALARA principles in the evaluation of the lessons learned from these events and how is NRC proposing to promote improvements to the design and operation of future NPPs with respect to prevention or mitigation of ground water contamination through leaking piping (also considering the ALARA principle)?

Answer: Since 1997, the NRC has had regulatory requirements established in 10 CFR 20.1406, "Minimization of Contamination." At that time,10 CFR 20.1406(a) and (b) provided the regulatory requirements for new applicants for licenses to describe how the proposed facility design and procedures for operation would minimize, to the extent practical, contamination of the facility and the environment. The NRC provided guidance on acceptable methods of implementing this regulation in RG 4.21, “Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning.”

Additional information for applicants is provided in NUREG-0800 in Section 11.2, “Liquid Waste Management System” (ADAMS Accession No. ML100740449) and Section 2.4.13, “Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters” (ML12191A330). Further guidance is given in Branch Technical Position (BTP) 11-6 (ML12191A325), which provides guidance in modeling and assessing radiological impacts from a potential release of radioactive materials contained in liquid radwaste tanks.

For applicants for design certifications or new combined licenses, the NRC has provided interim staff guidance on the design features and operation of SSCs that contain or handle radioactive material in Design Certification/Combined License Interim Staff Guidance (DC/COL-ISG-)6, “Evaluation and Acceptance Criteria for 10 CFR 20.1406 to Support Design Certification and Combined License Applications” (ML092470100).

On June 17, 2011, the NRC staff promulgated a new rule in 10 CFR 20.1406(c), which extended the regulatory requirements (previously applicable only to new applicants) to all licensees, including currently operating facilities. The new requirement is for licensees to conduct operations, to the extent practical, in ways that minimize the introduction of residual radioactivity into the site, including the subsurface. As part of this new rulemaking, Subpart F, “Surveys and Monitoring,” of 10 CFR 20 was also revised at 10 CFR 20.1501(a) to require that radiological surveys be made, including of the subsurface, that are reasonable under the circumstances to evaluate the concentrations or quantities of residual radioactivity and the potential radiological hazards of the residual radioactivity detected. Guidance on implementing this new requirement was provided in RG 4.22, “Decommissioning Planning During Operations” (December 2012).

With respect to prompt remediation of residual radioactivity (i.e., contamination), the Commission directed the NRC staff in SRM-SECY-07-0177, “Proposed Rule: Decommissioning Planning (10 CFR Parts 20, 30, 40, 50, 70, and 72; RIN: 3150-AH45),” to make further improvements to the decommissioning planning process by addressing remediation of residual radioactivity during the operational phase. The staff addressed the merits of a potential prompt remediation regulation in its response to the Commission in SECY-13-0108, “Staff Recommendations for Addressing Remediation of Residual Radioactivity during Operations.” The staff recommended that before conducting any potential rulemaking, staff should collect operating experience on the effectiveness of the existing decommissioning-planning rule (i.e., the Decommissioning Planning Rule (10 CFR 20.1406 and 20.1501)). The NRC staff proposed collecting operating data for calendar years 2013 through 2014, which would be sufficient information on which to base a staff recommendation on the need for additional regulation or guidance for prompt remediation.

In SRM-SECY-13-0108, dated December 20, 2013, the Commission approved the staff’s recommendation; i.e., the Commission directed staff to collect two years of data following implementation of the Decommissioning Planning Rule, engage stakeholders in a public meeting focused on operational experience, and give the Commission the staff's recommendation for addressing remediation of residual radioactivity during the operational phase of the facility.

Other references of interest include:

- IN 2012-05, " Abnormal Releases of Radioactive Material in Liquids Potentially Resulting In Groundwater Contamination" (ML120410213)
- NUREG/CR-7029, "Lessons Learned in Detecting, Monitoring, Modeling and Remediating Radioactive Ground-Water Contamination" (ML11118A087)

Further information on buried-piping activities is available at the following publicly available Web site: http://www.nrc.gov/reactors/operating/ops-experience/buried-piping-activities.html.

Question No. 196

1. Please explain the processes and goals to analyze Environmental Costs and Benefit.
2. Design objectives for single power plant are presented at Appendix I to 10CFR50 during normal operation. The ICRP recommends effective dose limit to public and radiation workers except hands, skin and lens. Please provide the plan to change design objectives based on ICRP 60 or 103 including specified radiation source targets and thyroid equivalent dose limit, etc.
3. Please provide the dose limits (or risk limits) and assessment points (site boundary or predetermined point) to protect public in case of severe accidents.

Answer:

1. The NRC staff performs two different types of environmental impact analyses for power reactors: one for applications to construct and/or operate a new reactor, and the second for applications to extend an existing operating license for an additional period of time, typically 20 years. Benefit cost analyses are handled differently in each of these situations.

   All environmental impact work is governed by the National Environmental Policy Act of 1969 (NEPA), which serves as the primary input for two NRC guidance documents: NUREG-1555, "Standard Review Plans for Environmental Reviews for Nuclear Power Plants" (the environmental standard review plans (ESRPs) for new reactors), and NUREG-1555, Supplement 1, "Standard Review Plans for Environmental Reviews for Nuclear Power Plants, Supplement 1: Operating License Renewal," (the ESRPs for license renewal). The two guidance documents serve as the NRC staff's primary source for the level of detail that is sufficient for a power reactor's environmental assessment. While NEPA allows environmental impact assessments to be based on reliable data obtained from external analyses (e.g., environmental assessments made by the applicant, NRC staff, or other entities), the processes are different for new and existing power reactors, especially in regard to benefit cost assessments.

   NEPA does not require a full monetized analysis of the benefits and costs (public and private) for a decision to be made on any action. In fact, the Council on Environmental Quality guidance for NEPA analyses repeatedly speaks of "straightforward and concise reviews" that are "...proportionate to potential impacts and effectively convey the relevant considerations in a timely manner to the public and decision makers... ." Consequently, the ESRP allows the use of "reasonably detailed information about economic benefits of the proposed action [needed] to assess any potential social or economic impacts that might occur," including net electrical generating benefits, any other commercial products produced by the proposed action, incremental increases in regional productivity, economic benefits (typically property taxes), and non-monetary benefits (e.g., new infrastructure or public services (ESRP Section 10.4.1)).

The 1996 version of the Generic Environmental Impact Statement (GEIS) states the Commission's position with regard to the role of benefit cost analysis: "The Commission has concluded that, for license renewal, the issues of need for power and utility economics should be reserved for State and utility officials to decide. Accordingly, the NRC will not conduct an analysis of these issues in the context of license renewal or perform traditional cost-benefit balancing in license renewal NEPA reviews" (GEIS Section 1.7.1, 1993). Additionally, as stated in Section 1.7.3 of the 1996 GEIS, "[t]he issues of need for power, the economic costs and benefits of the proposed action are specifically excluded from consideration in the supplemental environmental impact statement for license renewal by 10 CFR 51.95(c), except as these costs and benefits are either essential for a determination regarding the inclusion of an alternative in the range of alternatives considered or relevant to mitigation." Consequently, an environmental impact statement for the renewal of an existing operating permit will have limited (if any) benefit cost discussion.

However, economic costs and benefits are evaluated as components of the Severe-Accident Mitigation Alternative (SAMA) review. The SAMA review is an evaluation of alternatives to mitigate severe accidents. NRC staff evaluates SAMAs to ensure that changes to the nuclear power plant that could improve severe-accident safety performance are identified and evaluated. Potential improvements could include hardware modifications, changes to procedures, and changes to training programs. The outcome of the SAMA analysis is a list of plant improvements that meet the criteria of being potentially cost-beneficial. The only changes that the applicant commits to implement as part of the license-renewal process are those that are identified as being potentially cost-beneficial and that are related to adequately managing the effects of aging during the PEO (NUREG-1850, "Frequently Asked Questions on License Renewal of Nuclear Power Reactors").

In preparing SAMA analyses, license-renewal applicants are guided by the methodology presented in NUREG/BR-0184, "Regulatory Analysis Technical Evaluation Handbook," issued January 1997, and the guidance provided in NEI 05-01, Revision A, "Severe Accident Mitigation Alternatives (SAMA) Analysis, Guidance Document," issued November 2005. NRC Staff procedures for conducting a SAMA review are presented in NUREG-1555, Supplement 1, Revision 1, "Standard Review Plans for Environmental Reviews for Nuclear Power Plants: Environmental Standard Review Plan for Operating License Renewal," issued July 2013, see Section 5.2, "Severe Accident Mitigation Alternatives."

While the processes are different, the analyses are used in the same manner between new reactors and license renewal applications. The NRC cannot "reject" a power-reactor application based solely on environmental considerations. NEPA law does not confer any additional powers or authority on a federal agency beyond their statutory obligations (NEPA 1969 Sec. 104 [42 U.S.C. § 4334]), and NRC's statutory obligations (found in the Atomic Energy Act of 1954 (as amended)) do not explicitly provide NRC staff with the authority to deny an application based on environmental factors. Instead, environmental assessments are considered holistically, as one of many action-related impacts affecting the Commission's final decision.

2. Under SRM-SECY-12-0064, the staff has initiated the planning and revision of regulations and guidance in Appendix I to 10 CFR 50. The objective of the rulemaking is to revise the underlying dosimetry concepts of the design objectives in Appendix I to

be consistent with those of current ICRP recommendations under ICRP Publication 103. The rulemaking for Appendix I to 10 CFR 50 is being synchronized with a parallel rulemaking in revising 10 CFR Part 20 dosimetry basis. Once revised, the implementation of Part 20 and Part 50, Appendix I regulations will use common radiation terminology and definitions and dose calculation methods.

The rulemaking for Appendix I to 10 CFR 50 will necessitate an extensive revision of the supporting guidance, starting with RG 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I" (and several other Division 1 and Division 4 RGs), two computer codes, several NUREG documents on the implementation of associated operational programs, and conforming changes to other documents, such as Chapter 11 of NUREG-0800, RG 1.206, and NUREG-1555. NRC staff is working on a project plan detailing the overall course of the rulemaking, including identifying expected critical events and milestones of the rulemaking; defining the interdependency of specific tasks with the Part 20 rulemaking effort; coordinating with other U.S. Federal agencies (e.g., EPA, DOE, and Department of Labor) in issuing standardized regulations; conducting public meetings and industry workshops in seeking input from stakeholders; preparing the regulatory basis for revising the rule and its guidance; developing statements of work for contractor support; identifying staff; and scoping out resources for contractor support in revising computer codes and RGs

At this time, NRC staff is finalizing the *Federal Register* Notice announcing the Part 50, Appendix I rulemaking effort and seeking input from members of the public and industry. The NRC is planning a series of public meetings and industry workshops, with the first one proposed for late June 2014. At this time, the notice is expected to be published in late April 2014.

3. An applicant for a license for a nuclear power plant is required to demonstrate that the dose consequences from a suite of design-basis accidents meet certain dose guidelines for an individual located at two defined distances from the facility (i.e., the exclusion area boundary and the outer boundary of the low population zone). However, there are no explicit dose limits or assessment points established in NRC regulations for severe accidents. The following discussion provides more detail on the regulatory requirements, including those for the assessment, and associated regulatory guidance.

   That said, each nuclear power plant is required to develop, maintain, and follow an emergency plan that meets the regulations in 10 CFR 50.47 and Appendix E to 10 CFR Part 50. These plans are required to demonstrate that adequate protective measures can and will be taken to protect the public within a 10-mile-radius plume-exposure emergency planning zone in the event of a radiological emergency. These emergency plans are activated progressively through four emergency classifications, with public protective actions necessary at the General Emergency level. Nuclear power plant operators are required to provide a recommendation for public protective actions to State and local governments on declaration of a General Emergency. As a participant in the Federal National Response Framework, the NRC uses guidance provided to State and local governments by the EPA in its "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," EPA 400-R-92-001. This guidance states that protective actions such as evacuation or (in some cases) sheltering should be initiated when the sum of the effective dose equivalent and the committed effective dose equivalent at any location within the 10-mile emergency planning zone reaches 1 rem.

There are no explicitly identified offsite assessment points. However, emergency plans are required to provide dose-projection capabilities and offsite field-monitoring capabilities within the boundary of the 10-mile emergency planning zone. The protective action guides are assessed at the site boundary and beyond.

Subpart B, “Evaluation Factors for Stationary Power Reactor Site Applications on or after January 10, 1997,” of 10 CFR Part 100, “Reactor Site Criteria,” is the regulation used in the United States to evaluate the siting of new power-reactor combined-license and early site permit applications. 10 CFR 100.2(c)(2) requires that the site’s atmospheric dispersion characteristics are such that the radiological dose consequences of postulated accidents meet the criteria in 10 CFR 50.34(a)(1). 10 CFR 50.34(a)(1) requires that the application provide a safety assessment of the site and a safety assessment of the facility to be operated at the location. The safety assessment of the facility will include consideration of the engineered safety features, with special attention to those plant design features that are intended to mitigate the radiological consequences of accidents. 10 CFR 50.34(a)(1)(ii)(D) states that the assessment shall:

“…assume a fission product release[6] [see below] from the core into the containment assuming that the facility is operated at the ultimate power level contemplated. The applicant shall perform an evaluation and analysis of the postulated fission product release, using the expected demonstrable containment leak rate and any fission product cleanup systems intended to mitigate the consequences of the accidents, together with applicable site characteristics, including site meteorology, to evaluate the offsite radiological consequences. Site characteristics must comply with part 100 of this chapter. The evaluation must determine that:

“(1) An individual located at any point on the boundary of the exclusion area for any 2 hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 25 rem total effective dose equivalent (TEDE).

“(2) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a radiation dose in excess of 25 rem total effective dose equivalent (TEDE)”

Footnote 6 further defines the fission product release to be assessed:

“The fission product release assumed for this evaluation should be based on a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release into the containment of appreciable quantities of fission products.”

In practice, as discussed in NRC guidance documents, the postulated accident fission product release used in this assessment is based on modeling of severe accidents.

It should be noted that the dose criteria are not dose limits to the public, but are evaluation factors to determine acceptable siting of a nuclear facility. In fact, Footnote 7 states:

“A whole body dose of 25 rem has been stated to correspond numerically to the once in a lifetime accidental or emergency dose for radiation workers which, according to NCRP recommendations at the time could be disregarded in the determination of their radiation exposure status (see NBS Handbook 69 dated June 5, 1959). However, its use is not intended to imply that this number constitutes an acceptable limit for an emergency dose to the public under accident conditions. Rather, this dose value has been set forth in this section as a reference value, which can be used in the evaluation of plant design features with respect to postulated reactor accidents, in order to assure that such designs provide assurance of low risk of public exposure to radiation, in the event of such accidents.”

The definitions of “exclusion area” and “low population zone” are given in 10 CFR 50.2. “Definitions.” The location of the exclusion area’s boundary is defined by the applicant and might be nearer to the facility than the demarcated edge of the facility property (i.e., the fence line), at a more distant location, or co-located with the fence. Some applications might call this distance the “site boundary” in documentation for the dose calculations. In general, the exclusion area’s boundary is a circle inscribing a calculated distance from the facility’s center point. Many recent applications have defined the exclusion area boundary as 0.5 mi (0.8 km). The outer boundary of the low-population zone is also defined by the applicant and has generally been 1 to 2 miles (1.61 to 3.22 km) distant from the center point of the facility.

10 CFR Part 52 also requires that the same design-basis accident dose-analysis requirements be demonstrated in the safety analysis report included in the application for early site permits (10 CFR 52.17(a)(1)(ix)), standard design certifications (10 CFR 52.47(a)(2)(iv)), combined licenses (10 CFR 52.79(a)(1)(vi)), standard design approvals (10 CFR 52.137(a)(2)(iv)), and manufacturing licenses (10 CFR 52.157(d)).

To perform review of light-water reactor design and site applications, the NRC staff uses the guidance in Section 15.0.3, “Design Basis Accident Radiological Consequences of Analyses for Advanced Light Water Reactors,” of NUREG-0800. Guidance to applicants and licensees on design-basis accident dose analyses for new applications for large light-water reactors is given in RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.” Both of these documents and the regulations discussed above are available on the NRC’s Web site.

In contrast to the design-basis accident analyses discussed above, there are no regulatory dose limits or criteria for severe accidents. The risk from operation of the facility is evaluated in a PRA for the facility. Section 19.0, “Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors,” of NUREG-0800 provides guidance on staff review of nuclear power-plant risk in new applications. In 1986, the NRC issued a policy statement on Safety Goals for the Operation of Nuclear Power Plants (at 51 FR 28044 (August 4, 1986) as corrected and republished at 51 FR 30028 (August 21, 1986)).

Question No. 197
At the EPA in 40CFR190, dose constraints for multi-unit nuclear power plants are presented. Please explain how to determine the discharge point by single unit or representative discharge point (distances and meteorological data), and then how to estimate the assessment point (site boundary or predetermined point) from multi-units.
<u>Answer</u>: The NRC does not have specific regulatory guidance on the question of estimating the discharge point from multi-unit facilities. However, for calculation of population doses, RG 1.109 (see below) defines the distance based on the "center of facility" (Section 2.b of Appendix D). Also, Chapter 3 of NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants: A Guidance Manual for Users of Standard Technical Specifications," addresses special considerations for multi-unit sites. NUREG-0543, "Methods for Demonstrating LWR Compliance with the EPA Uranium Fuel Cycle Standard (40 CFR Part 190)" (ADAMS Accession No. ML081360410), provides specific guidance on compliance with the EPA regulations in 40 CFR 190, "Environmental Radiation Protection Standards for Nuclear Power Operations." The NUREG concludes that there is reasonable assurance that sites with up to four operating reactors that have releases within Appendix I design-objective values are also in conformance to the Uranium Fuel Cycle Standard (40 CFR Part 190), provided that annual exposure resulting from direct radiation (e.g., from store components such as an independent spent-fuel storage installation, reactor vessel heads, steam generators) is below about 1 mrem. The NRC under 10 CFR 20.1301(e) requires licensees to comply with 40 CFR 190, but does not have specific regulatory guidance on the question of estimating the discharge point from multi-unit facilities and aggregate inventories of radioactive materials released over the entire fuel cycle as a function of electrical energy produced by power reactors. This requirement is also described in GL 79-041, which mandates that reactor licensees establish methods to demonstrate compliance with EPA regulations. However, for calculation of population doses, RG 1.109 (see below) defines the distance based on the "center of facility" (Section 2.b of Appendix D). NRC references related to this question are as follows: 1. Section 2.b, "Population-Integrated Doses from Airborne Effluents," of Appendix D, "Models for Calculating Population Doses from Nuclear Power Plant Effluents," to Revision 1 of RG 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I" (ADAMS Accession No. ML003740384). 2. Revision 2 of RG 1.21, "Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste" (ML091170109), - Section C.1.2, "Release Points for Effluent Monitoring," and - Section C.3, "Effluent Dispersion (Meteorology and Hydrology)." 3. Revision 1 of RG 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors" (ML003740354). 4. Revision 1 of RG 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I" (ML003740390). 5. NUREG-0543, "Methods for Demonstrating LWR Compliance With the EPA Uranium Fuel Cycle Standard (40 CFR Part 190)" (ML081360410).

Question No. 214
15.1 Authorities and Principles It is mentioned that the Commission disapproved any change to the occupational limit for effective dose. Please describe the main reason.
Answer: The current limit on occupational total effective dose equivalent is 50 mSv (5 rem) per year. This value is the same as the single-year value recommended by the ICRP. Regulatory requirements in the United States, including those of the NRC, must be completed through a rulemaking process which includes notice for public comment. A rulemaking must include a regulatory analysis of benefits and impacts of the proposed action, and, in addition, a "backfit" analysis as required in 10 CFR 50.109; 10 CFR 52.39, 52.63, 52.83, 52.98, 52.145, and 52.171; Appendices A, B, and C, and Section VIII of Appendix D to 10 CFR 52; 54.37; 10 CFR 70.72; 10 CFR 72.62; and 10 CFR 76.76 (for titles, see http://www.nrc.gov/reading-rm/doc-collections/cfr/). A backfit analysis requires that there be an assessment of benefits and impacts, and that the proposal is justified by those benefits, unless the requirement is considered as necessary for adequate protection. The NRC staff continues with its development of a regulatory basis for possible changes in ways consistent with Commission direction. The NRC staff, in SECY-12-0064, had proposed to the Commission that it continue to explore the implications of lowering the dose limit. During the Commission's deliberations, the ACRS also reviewed the subject; provided a letter to the Commission, dated October 16, 2012; and concluded that rulemaking to revise the limits for occupational radiation exposure should not be undertaken. That letter stated that a decision to change dose limits should be based on demonstrated health and safety benefits and consideration of negative, unintended safety consequences. The letter stated that there is disagreement among professional organizations involved in radiation protection regarding the health and safety benefits of reducing the dose limit from 50 mSv (5 rem) to 20 mSv (2 rem) per year. In the absence of a clear and well-demonstrated benefit, the ACRS disagreed with lowering the dose limit. The ACRS also stated that compliance with lower dose limits could also have unintended negative consequences. Finally, the ACRS noted the excellent performance of the nuclear power industry in implementing the ALARA principle, which has resulted in exposures well within the national and international recommendations. Given the industry performance, the ACRS recommended that the staff develop improvements in the NRC guidance for more effective implementation of ALARA strategies and programs in those segments of the regulated community that have identified compliance problems. In summary, because exposures in the nuclear power industry have been reduced through ALARA implementation, a change in the limits would not substantially change the actual exposure of occupational workers, and could entail potentially significant costs and negative consequences. The Commission, in SRM-SECY-12-0064, disapproved the staff's recommendation to develop the regulatory basis to reduce the occupational total effective dose equivalent. The SRM also directed the staff to continue discussions with stakeholders on alternative approaches to deal with individual protection at or near the current dose limit. The Commission approved the staff developing the regulatory basis to adopt the updated methodology and terminology of ICRP Publication 103. The NRC staff is currently preparing an Advance Notice of Proposed Rulemaking to specifically solicit further public input on the issues of updating methodology and terminology, ALARA implementation, and possible additions to ALARA requirements in response to the Commission's direction.

ARTICLE 16. EMERGENCY PREPAREDNESS

(i) **Each Contracting Party shall take the appropriate steps to ensure that there are onsite and offsite emergency plans that are routinely tested for nuclear installations, and cover the activities to be carried out in the event of an emergency.**

(ii) **For any new nuclear installation, such plans shall be prepared and tested before it [the installation] commences operation above a low power level agreed [to] by the regulatory body.**

(iii) **Each Contracting Party shall take appropriate steps to ensure that, insofar as they are likely to be affected by a radiological emergency, its own population and the competent authorities of the States in the vicinity of the nuclear installation are provided with appropriate information for emergency planning and response.**

(iv) **Contracting Parties that do not have a nuclear installation on their territory, insofar as they are likely to be affected in the event of a radiological emergency at a nuclear installation in the vicinity, shall take the appropriate steps for the preparation and testing of emergency plans for their territory that cover the activities to be carried out in the event of such an emergency.**

This section discusses (1) the background of emergency planning in the United States, (2) offsite emergency planning and preparedness, (3) emergency classification system and emergency action levels, (4) recommendations for protective action in severe accidents, (5) inspection practices and regulatory oversight, (6) response to an emergency, (7) communications with neighboring states and international arrangements, (8) communications with the public, and (9) lessons learned from the Fukushima event.

Question No. 129
The USA has agreements with its neighbours for nuclear emergencies. What joint emergency exercises have been held with Canada or Mexico (or are planned) to test these arrangements?
Answer: In 2013, representatives from the Canadian Nuclear Safety Commission observed an emergency-preparedness exercise at a U.S. nuclear power plant and NRC representatives observed an emergency-preparedness exercise at a Canadian nuclear power plant. The NRC and the Canadian Nuclear Safety Commission have ongoing dialogue regarding participation in exercises in the future. Additionally, communications with the NRC's Canadian and Mexican counterparts are tested regularly during domestic emergency-preparedness exercises conducted by the NRC's emergency-response personnel.
Question No. 130
Please give an overview of the plans in place for responding to a nuclear accident occurring in a neighbouring country or elsewhere that impacts on the USA or its interests overseas. How has the experience of responding to the Fukushima accident influenced or changed these plans?
Answer: The EPA is the lead U.S. Government agency for coordinating Federal efforts for accidents involving foreign, unknown, or unlicensed radioactive materials. The "United States-Canada Joint Radiological Emergency Response Plan" ("Joint Plan") was signed on April 27, 1996, in Vancouver, British Columbia. The Joint Plan establishes the basis for a

cooperative response to peacetime radiological events that involve both countries. It also applies when either country needs the other's help in responding to a potential or actual radiological event on its own soil.

The Joint Plan was developed under the umbrella of the "Agreement between the Government of Canada and the Government of the United States of America on Cooperation in Comprehensive Civil Emergency Planning and Management (1986)." The U.S./Canada Radiological Emergency Preparedness Group, of which the EPA is a member, maintains and revises it.

The Joint Plan contains procedures for critical aspects of a joint response:

- alerting the appropriate federal authorities in each country to an actual or potential radiological threat,
- establishing cooperative measures to minimize the threat to public health and safety, property, and the environment, and
- facilitating coordination among Federal organizations in both countries as they provide support to affected States and Provinces.

In addition, the bilateral agreements with Canada and Mexico allow the exchange of technical staff and the sharing of certain technical information. The U.S. Government is currently developing a protocol for responding to international chemical, biological, radiological, and nuclear incidents. This protocol will describe the roles and responsibilities of U.S. Government departments and agencies in the event of a chemical, biological, radiological, or nuclear incident outside of the United States and will incorporate the lessons learned from the response to the Fukushima accident.

Depending on incident specifics, the NRC could use established relationships with international regulatory counterparts to facilitate incident-related communication and provide technical advice and assistance to the affected country.

Industry Actions: Since the Fukushima accident, the commercial nuclear power industry in the United States has established procedures and agreements to share information, equipment, and personnel if needed during an event at a nuclear power plant. An Industry Response Center has been established at INPO's offices in Atlanta, Georgia, to coordinate information exchange and logistics. Discussions are underway with plant operating organizations in Canada and Mexico to arrange for information sharing and support during a nuclear-plant event in either country.

WANO also recognized the need to improve its ability to coordinate worldwide assistance to a plant experiencing a significant event. They are establishing a coordination center in their London offices. The center will have the capability to receive information on event progression and assistance needs from the affected organization and transmit it to other plant operators around the world. For an event occurring in the United States, the WANO liaison individual in the Industry Response Center at INPO will provide up-to-date information to London for communication to others. For an event outside the United States, the Industry Response Center will receive information and assistance requests from WANO and will coordinate dissemination and response from the United States.

The Industry Response Center protocols during an event include establishing direct communications with the NRC's Operations Center by telephone and, potentially, by sharing emergency-management actions directly using a software tool that is used by both organizations. If assistance from the U.S. Government were required (such as to support shipment of equipment to another country), personnel in the Industry Response Center would work through the NRC and/or the U.S. State Department to expedite actions.

Question No. 131

NUREG-0654 is being revised to align with the NRC emergency preparedness rule changes, which became effective in December 2011, and with the revised FEMA Radiological Emergency Preparedness Program manual issued in 2011.

Question is: are the 2011 revision includes lessons learned from Fukushima Accident? What are the main points?

Answer: The NRC's emergency-preparedness regulations were revised and issued in late 2011 as part of a multi-year effort that began well before the Fukushima Dai-ichi event. These regulations might be further revised in response to NRC's Fukushima lessons-learned initiative, but to date no new emergency-preparedness rule changes have been proposed. The NRC and FEMA are currently preparing a revision to the implementing guidance for the updated emergency-preparedness regulations, specifically NUREG-0654/FEMA-REP-1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants." The guidance is based on the 2011 rule changes; however, because the regulations have not been changed for Fukushima, corresponding guidance changes are not within the current scope.

Question No. 132

In your report it says:
Addressees also were asked to assess both their current staffing levels and the appropriate staff and positions to respond to a multi-unit event given a beyond-design-basis natural event and to determine if enhancements were needed. ....
On April 30, 2013, licensees provided a staffing assessment to respond to the first phase of the RFI. The staff is currently reviewing these submittals and expects to issue the results of the staff's review no later than December 2013. ....

Could you explain the responses of licensees and the NRC review results for these?
Especially could you explain of enhancement in the area that the NRC thinks good practices?

Answer: The NRC staff has completed the reviews of the staffing assessments provided by U.S. licensees in response to letters issued on March 12, 2012. The licensees assessed whether the minimum staffing as outlined in their currently approved emergency plans was sufficient to respond to and implement the station-blackout mitigation strategies in place at the time the letters were issued. The NRC review of the submitted staffing assessments determined that all licensees could implement the initial portion of the assessment for station-blackout strategies with their minimum staffing as contained in their currently approved emergency plans. The next portion of the staffing assessment will provide an evaluation based on the mitigation strategies developed in response to NRC Order EA-12-049, "Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events." This NRC Order will require licensees to meet certain requirements and to develop mitigation strategies that could involve such items as the deployment and operation of portable equipment around the site in response to an event using augmented emergency-response personnel. The NRC reviews were in the context of necessary regulatory standards, but not in the context of good practices. The U.S. industry participates in industry organizations (e.g., INPO) that review these matters from a good-practices perspective.

Question No. 133
The report indicates a discussion whether potassium iodine should be prestaged beyond the current 10 mile zone. However in case of a non-filtered accidental release (NPP's do not seem be equipped with adequate filters), the need for iodine blocking as protective action above the 10 miles is most likely. Could a centralized stockpiling of stable iodine above the 10 miles zone be an option for achieving a kind of a graded approach above the 10 miles?
Answer: The NRC identified (in its Near-Term Task Force report on the events at Fukushima) the pre-staging of potassium iodide beyond the current 10-mile plume-exposure emergency planning zone as an issue for consideration. This issue has been classified as a Tier 3 (lower-priority) issue for future review. SECY-13-0095 presents the status of this effort, which has not begun.
Question No. 134
SAMGs have been developed (by nuclear industry) in the USA and later also implemented in many other countries. Why do SAMGs not have a legal base in US regulations?
Answer: In GL 88-20, Supplement 2, "Accident Management Strategies for Consideration in the Individual Plant Examination Process," dated April 4, 1990, the NRC accepted industry commitments to develop and implement SAMGs but did not incorporate SAMGs into its regulatory requirements. This approach was part of an overall plan that included other activities such as the issuance of the NRC's Severe-Accident Policy Statement and the completion of the individual plant examinations. In light of the Fukushima accident, the NRC is revisiting the possible addition of regulatory requirements related to SAMGs.
Question No. 135
What are the dose criteria for protective measures in the US and are they in line with the International Basic Safety Standards (ICRP 103)?
Answer: The dose criteria guidelines for protective measures in the United States are based on ICRP 60 and are consistent with ICRP 103 recommendations. The protective action guidelines for sheltering and evacuation in the United States are 10- to 50-mSv total effective dose equivalent and 50-mSv child thyroid dose for the recommendation of potassium iodide based on the dose in the first 4 days. These dose criteria and guidelines are consistent with the ICRP 103 guidelines, which establish a range of 20- to 100-mSv effective dose and 50-mSv thyroid dose based on the dose in the first 7 days.
Question No. 136
Could you please specify communication equipment improvements in case of extensive damage in the surroundings of the plant, that could affect to normal and emergency systems, including cellulars, for example, to contact with on call shift?
Answer: In response to letters issued to NRC licensees on March 12, 2012, U.S. licensees proposed interim actions (the use of portable satellite phones) and long-term enhancements (the use of new radio systems, sound-powered telephones, battery-operated repeaters, and satellite phone systems) to ensure that they could effectively communicate during a station-blackout event. U.S. licensees also have in place a "self-activation" process in which the site emergency-response organization would report to predetermined response facilities in the event of a beyond-design-basis external event that impacted the communications infrastructure and the site's ability to contact the needed responders. The NRC staff determined that the interim actions and long-term action enhancements proposed by U.S. licensees were acceptable methods to ensure that communications would be maintained throughout the postulated event.

Question No. 137

In the last paragraphs of section 16.4 the report states that the NRC considers evacuation and sheltering to be the two primary protective actions and that thyroid-blocking potassium iodide is a supplementary protective action. Also, the report states that NRC's amended regulation requires that each State consider giving potassium iodide to the general public as a protective measure, supplementing the evacuation and sheltering protective actions. Our questions are: 2) In the last paragraph of this section it states that the NRC's guidance on evacuation and sheltering is consistent with guidance in IAEA TECDOC's 953 and 955 both from 1997. Has the NRC considered using the updated documents issued by the IAEA in the last few years (e.g., Safety Series Guide No. GSG-2) or, alternatively or in conjunction, basing the radiological protection standards on the more recent ICRP 103 and associated publications that followed from ICRP that specifically address radiological protection in emergency situations(ICRP 109 and 111)?

Answer: The EPA is the agency responsible for establishing appropriate radiation-protection guidance for the United States, including establishment of protective-action guidance for nuclear power-plant and other radiological and nuclear accidents. The NRC works closely with the EPA and other Federal agencies in the development of protective-action strategies and dose limits and guidelines. The EPA recently updated its protective-action guidelines manual to include ICRP standards (such as ICRP 60, "1990 Recommendations of the International Commission on Radiological Protection") and the NRC has similarly updated its dose-projection software (Radiological Assessment System for Consequence Analysis (RASCAL)) based on ICRP 60. The protective-action guideline recommendations in the EPA protective-action guidelines manual remain in the range of a 10-to 50-mSv total effective dose equivalent for sheltering and evacuation recommendations and at a 50-mSv child thyroid dose for the recommendation of potassium iodide based on the dose in the first 4 days. This guideline is consistent with that of ICRP 103, which establishes a range of a 20- to 100-mSv effective dose and a 50-mSv thyroid dose based on the dose in the first 7 days.

Question No. 138

It is reported that to enhance emergency preparedness post Fukushima event, communication assessment were carried out by licensee and reported to USNRC on October 31, 2012. Based on these, USNRC have identified some generic issues. Can USA provide more details on the generic issues found related to communication assessment?

Answer: The NRC staff identified eight generic technical issues related to communications based on licensee communication submittals. These generic technical issues included (1) the availability of power for analyzed equipment and the maintenance of communications enhancements, (2) a description of the use and function of communication enhancements, (3) the protective storage of both new and existing communications enhancements and equipment in the event of a large-scale natural event, (4) programmatic control for the use of new equipment purchased as a planned enhancement, (5) a description of the assumptions used for the licensee's communications assessment, (6) the notification of plant personnel in the event of a large-scale natural event which resulted in the loss of alternating-current power, (7) the maintenance of plant communications when communication enhancements are being implemented, and (8) a description of how communication would be maintained with onsite and in-plant response teams and offsite response organizations if their communication links were unavailable. These generic issues were discussed and resolved with U.S. licensees during a series of public meetings conducted by the NRC.

Question No. 139

There is a requirement for the offsite response agencies and the State to participate in regular emergency exercises. What level of involvement do the Federal agencies, such as NRC and

FEMA, have in these exercises? How often is the national (Federal Government) response required to be exercised?
Answer: The NRC's primary roles in response to an event at a U.S. licensed facility are to provide oversight of the licensee's actions to mitigate the event; validate the licensee's Protective Action Recommendations made to the State; and communicate with the public, Federal and State Governments, members of Congress, officials at the White House, and international partners. Those responsibilities are demonstrated through participation in exercises with the licensed facility. The NRC's four Regional Offices participate with each facility within their respective NRC Region once every six years. During exercises, the Regions activate their Incident Response Center, assemble a response team, and deploy a team to co-locate onsite. NRC Headquarters participates in one of these exercises per Region per year. The requirement for the NRC's participation in exercises resides within the agency's Incident Response Program. FEMA provides Federal oversight of the State, local and Tribal governments for radiological emergency-preparedness exercises. This includes involvement in the planning and execution of, as well as an in-depth evaluation of, each biennial exercise. The Department of Homeland Security and FEMA practice Federal response on a regular basis, to include a Capstone Exercise (National Level Exercise) every two years as part of the National Exercise Program. The Federal Government does not have a specific requirement to exercise the full Federal response force in support of a radiological incident; however, a number of the National Level Exercises have been focused on a radiological event. In 2015, an exercise with full Federal participation is planned with a nuclear power-plant scenario as the focus.
Question No. 140
It should be the task of the regulatory body to provide timely and independent information to the public in case of an incident, accident or emergency. How is NRC fulfilling this requirement?
Answer: The U.S. NRC maintains a robust crisis-communication program that provides accurate and timely incident information through a variety of communication tools. The comprehensive crisis-communication strategy is tested regularly with emergency-preparedness exercises and real events and is updated as necessary. The strategy outlines staffing, messaging, crisis-communication philosophies and prewritten press materials. The strategy uses both traditional media-relations tools and direct-to-the-public communication through the agency Web site and social-media platforms. It is also integrated into the Federal government framework for emergency response.
Question No. 141
In the article 16 of the report, it is mentioned that mitigating strategy adresses periodic training and exercise for prolonged SBO scenario. Can USA clarify requirement on duration for which prolonged SBO to be considered and elaborate the basis for the same?
Answer: The NRC issued a mitigation-strategies order on March 12, 2012, requiring all U.S. nuclear power plants to implement strategies that will allow them to cope without their permanent electrical power sources for an indefinite amount of time. These strategies address keeping the reactor core and spent fuel cool, as well as protecting the thick concrete containment buildings that surround each reactor. The Order requires a three-phase approach for mitigating beyond-design-basis external events. The initial phase requires the use of installed equipment and resources to maintain or restore key safety functions, including core cooling, containment, and spent fuel pool cooling. The transition phase requires providing sufficient, portable, onsite equipment and consumables to maintain or restore these functions until they can be accomplished with resources brought from

offsite. The final phase requires obtaining sufficient offsite resources to sustain those functions indefinitely.

Fukushima has demonstrated the necessity for nuclear power plants to have indefinite coping capability during a beyond-design-basis event.

Question No. 142

In the last paragraphs of section 16.4 the report states that the NRC considers evacuation and sheltering to be the two primary protective actions and that thyroid-blocking potassium iodide is a supplementary protective action. Also, the report states that NRC's amended regulation requires that each State consider giving potassium iodide to the general public as a protective measure, supplementing the evacuation and sheltering protective actions. Our questions are:

1) Based on which principles of radiological protection is thyroid-blocking using potassium iodide a supplementary protective action instead of an automatic action based on a trigger? That is to say, how is it justified to not have a stricter procedure for thyroid-blocking?

Answer: Initial protective-action decisions such as those involving sheltering and evacuation are made based on plant conditions. The emergency action levels developed by each nuclear power plant enable the plant operator to evaluate the increasing seriousness of events unfolding at the plant and to develop protective-action recommendations for transmittal to offsite authorities before a release occurs. The timeliness of these actions enables State and local response officials to order citizens to evacuate or shelter (if appropriate) before a release of radioactive materials in excess of technical specifications. Making protective-action decisions based on escalating plant conditions rather than waiting to make such decisions until a release occurs enables maximum dose savings to the whole body (the ALARA principle). Potassium iodide provides protection (1) for only one organ and (2) only from internal exposure and (3) only if the potassium iodide is taken at the appropriate dose and time. Because the NRC and FEMA do not require potassium iodide to be a part of States' emergency plans (though it is required to be considered), implementing Federal requirements for its administration is not appropriate.

Question No. 143

In the last paragraphs of section 16.4 the report states that the NRC considers evacuation and sheltering to be the two primary protective actions and that thyroid-blocking potassium iodide is a supplementary protective action. Also, the report states that NRC's amended regulation requires that each State consider giving potassium iodide to the general public as a protective measure, supplementing the evacuation and sheltering protective actions. Our questions are: Regarding the text in the report referred to above: NRC's amended regulation "requires that each State consider giving potassium iodide to the general public as a protective measure, supplementing the evacuation and sheltering protective actions". This implies that different States can have different decisions regarding giving potassium iodide to the general public, since it is stated that "each State consider". In the event of an accident that would cause enough radioactive iodine emissions to motivate iodine tablets as a protective action, and, at the same time more than one State is affected, have the crisis communication plans dealt with the problem of how to inform the public that different States have different levels of radiological protection?

Answer: Each State or local authority makes its own decisions regarding protective actions for nuclear power-plant events following the guidelines in the EPA Protective Action Guidelines as well as appropriate NRC and FEMA guidance. Offsite emergency plans for nuclear power installations are evaluated by FEMA on an ongoing basis. Each State decides on the appropriateness of including potassium iodide in its emergency plans. States that share the

plume-exposure pathway emergency planning zone across state lines develop common protective-action strategies, including potassium iodide distribution. Such common plans ensure consistent crisis communications for the nuclear power-plant accident.
Question No. 144
In section 16.9 at the end of the article 16 text, on page 185, there is text on recommendations that require further staff study: emergency planning zones and whether potassium iodine should be prestaged beyond the current 10 mile zone. Please make clearer what the NRC has identified in its lessons learned, that is, what is being planned to be evaluated: distribution of potassium iodine beyond the current 10 mile zone, recommendation to take or recommendation to have in each house potassium iodine beyond the current 10 mile zone, or? What is the current recommendation is regarding potassium iodine within the 10 mile zone? Also, a definition of the word “prestaged” would be helpful.
Answer: NRC regulations require that a range of protective actions be developed for the 10-mile plume-exposure pathway emergency planning zone. U.S. licensees have the obligation to confirm that offsite authorities have considered the use of potassium iodide as a supplemental protective action for the general public. States and local jurisdictions which contain a nuclear power plant consider the use of potassium iodide, but the NRC cannot require specific distribution plans for potassium iodide. Not all States and local jurisdictions have included potassium iodide use in their emergency plans. The decisions on how to distribute potassium iodide are left up to the State and local response authorities. As a result of the accident at Fukushima, the NRC’s Near-Term Task Force recommended, and the staff is evaluating as a Tier 3 action, the appropriateness of the size of the 10-mile emergency planning zone as well as the need to expand the distribution of potassium iodide beyond the current 10-mile emergency planning zone. “Pre-staged” refers to potassium iodide tablets that have been purchased before an accident or incident; have been staged at reception or evacuation centers; and are held in State stockpiles for distribution to schools, hospitals, child-care facilities, and individual residents.
Question No. 145
Currently, SAMGs are voluntary and Recommendation 8 from the NRCs NTTF review of insights from the Fukushima accident addresses the integration of emergency operating procedures and SAMGs. When is the NRC expected to declare SAMGs and related training mandatory to all plant operators?
Answer: For the Emergency Onsite Response Capabilities rulemaking, initiated as a result of NTTF Recommendation 8 (i.e., strengthening and integration of emergency operating procedures, severe accident procedures (SAMGs), and extensive damage-mitigation guidelines), the NRC issued a draft regulatory basis for public comment on January 8, 2013. The staff is currently considering feedback from both internal and external stakeholders and modifying the document. The final rule, when complete, is expected to establish standards that ensure that plants can smoothly transition between various emergency procedures, keeping overall strategies coherent and comprehensive. The final rule is scheduled for completion in March 2016. This is discussed in Enclosure 1 to SECY 13-0095, http://www.nrc.gov/reading-rm/doc-collections/commission/secys/2013/2013-0095scy.pdf.

Question No. 146

For those States considering the use of potassium iodide, what is the prevailing strategy for its distribution in case of an accident?

Answer: Each State that includes potassium iodide in its emergency plan is also responsible for the potassium iodide distribution plan covering the population located within the 10-mile emergency planning zone. Some States with large transient populations decide to stockpile potassium iodide for post-accident distribution at designated centers; other States with smaller populations have distributed potassium iodide through the postal system. The most common type of distribution plan includes a combination of predistribution to citizens and schools and stockpiles in reception or other designated centers for distribution in the post-accident phase.

Question No. 147

Is the exchange of accident-related information between neighbouring countries trained during exercises?

Answer: Yes. The NRC has discussed the sharing of accident information with several other IAEA member states during bilateral discussions and during the observation of U.S. nuclear power-plant exercises by other member states. During every emergency exercise that NRC headquarters response teams have participated in since Fukushima, appropriate accident information has been developed using the IAEA Unified System for Information Exchange. Accident information stored in the Unified System for Information Exchange has not been transmitted to IAEA's Incident and Emergency Centre during nuclear power-plant exercises; however, the NRC will, in the future, begin to transmit exercise information to the Incident and Emergency Centre as a part of the exercise process.

Question No. 190

The National Report notes that the Federal response to a potential nuclear or radiological incident is designed to support the efforts of the facility operator and offsite officials. For such emergencies, Federal response activities are carried out in accordance with the National Response Framework's Nuclear/Radiological Incident Annex, which describes the roles of DHS, coordinating agencies (e.g., the NRC during an incident with one of its licensees), and other supporting Federal agencies. Please give more details about the responsibilities of FEMA within the framework of the National Incident Management System (NIMS) for different categories of emergencies including security incidents.

Answer: In simplest terms, during an emergency that involves a Federal response, FEMA fully activates the National Response Coordination Center whose purpose is to manage the emergency through various Emergency Support Functions. These Emergency Support Functions are performed by numerous Federal agencies. During Presidentially declared emergencies, FEMA manages the mission-assignment process that includes tasks to other Federal agencies. These tasks are often supported through disaster funding from Congress that FEMA manages.

Question No. 198

1. Please describe the composition of the Steering Committee for Emergency Planning and the administration process of the Committee including decision making system.

2. Please explain how to evaluate each respective agency's radiological emergency preparedness programs.

3. Please provide the examples for the development and implementation of proposed changes to radiological emergency preparedness-related regulations and guidance.

Answer:

1. The requirement for the Steering Committee is provided in a Memorandum of Understanding between the NRC and FEMA, which is contained in Appendix A, "Memorandum of Understanding Between Federal Emergency Management Agency and Nuclear Regulatory Commission," to 44 CFR 353, "Fee for Services in Support, Review and Approval of State and Local Government or Licensee Radiological Emergency Plans and Preparedness." In accordance with the memorandum, the Steering Committee was established to: (1) assure coordination of efforts to maintain and improve emergency planning and preparedness for nuclear power reactors based on the areas of cooperation outlined in the memorandum; and (2) coordinate the consistent development and application of criteria for licensee, State, and local emergency plans and preparedness. The Steering Committee consists of NRC managers and staff and is co-chaired by the NRC Director of the Division of Preparedness and Response in the Office of Nuclear Security and Incident Response and the FEMA Director of the Technological Hazards Division. The Steering Committee charter can be found on the NRC's Web site (ADAMS Accession No. ML091530139).

2. The NRC and FEMA radiological emergency-preparedness programs are evaluated through (a) internal and licensee feedback after exercises and self-assessment, (b) audits by the Office of Inspector General and the Government Accountability Program, and (c) programs such as the IAEA IRRS missions.

3. The emergency-preparedness regulations were amended in December 2011 and are summarized in the following 12 topics:

   On-Shift Staffing: The final rule requires licensees to perform a staffing analysis of on-shift personnel assigned emergency-response duties to ensure that these emergency responders do not become overburdened during an emergency event. This requirement is incorporated in 10 CFR Part 50, Appendix E, Section IV.A.

   Emergency Action Levels for Hostile Action: The final rule amends the regulations to require licensees to have Emergency Action Levels for events involving hostile action. This requirement is incorporated in 10 CFR Part 50, Appendix E, Section IV.B.

   Emergency-Response Organization Augmentation and Alternate Facilities: The final rule amends the regulations to require licensees to identify alternative facilities to support Emergency-Response Organization augmentation during hostile action. The rule codifies the Interim Compensatory Measures requirements associated with EA-02-026, "Order for Interim Safeguards and Security Compensatory Measures," and the enhancement examples described in Bulletin 05-02, "Emergency Preparedness and Response Actions for Security-Based Events." This requirement is incorporated in 10 CFR Part 50, Appendix E, Section IV.E.

   Licensee Coordination with Offsite Response Organizations During Hostile Action: The final rule amends the regulations to require licensees to identify, and provide a description of, the assistance expected from Offsite Response Organization personnel who would respond to the site during a hostile action. This requirement is incorporated in 10 CFR Part 50, Appendix E, Section IV.A.7.

   Protection for Onsite Personnel: The final rule amends the regulations to require specific emergency-plan provisions to protect onsite emergency responders, and other onsite personnel, in emergencies resulting from hostile action at nuclear power plants. This requirement is incorporated in 10 CFR Part 50, Appendix E, by creating a new Section I.

Challenging Drills and Exercises: The final rule amends the regulations to require licensees to include hostile action scenarios and other scenario variations in drills and exercises and to submit the scenarios for NRC review. The final rule also increases the exercise planning cycle from six to eight years to allow more flexibility in varying scenarios. Additionally, the ingestion exercise planning cycle is increased from six to eight years. These requirements are incorporated in 10 CFR Part 50, Appendix E, Section IV.F.

Backup Means for Alert and Notification Systems: The final rule amends the regulations to require that backup measures for the alert and notification system be available. The backup measures would be implemented if the primary means of alerting and notification were unavailable during an emergency. This requirement is incorporated in 10 CFR Part 50, Appendix E, Section IV.D.3.

Emergency Declaration Timeliness: The final rule amends the regulations to ensure that licensees have the capability to complete emergency declarations within 15 minutes in the event of a radiological emergency. This requirement is incorporated in 10 CFR Part 50, Appendix E, Section IV.C.

Emergency Operations Facility—Performance-Based Approach: The final rule amends the regulations to provide a performance-based criterion for emergency operations facilities. As such, licensees would be able to create emergency operations facilities servicing multiple sites if the criterion were met (commonly referred to as “combined emergency operations facilities”). The regulations were also revised to remove the references to an emergency operations facility as a “near site” facility and to incorporate specific emergency operations facility distance criteria in relation to a nuclear power plant site into the regulations. This would include providing a facility located near the site for use by NRC and offsite officials if the emergency operations facility was more than 25 miles away. These requirements are incorporated in 10 CFR 50.47(b)(3); 10 CFR 50.47(d)(1); 10 CFR 50.54(gg)(1)(i); and 10 CFR Part 50, Appendix E, Sections II., IV.E.8, IV.E.9.c, and IV.E.9.d.

Evacuation-Time Estimate Updating: The final rule amends the regulations to require licensees to periodically review and update evacuation-time estimates. These requirements are incorporated in 10 CFR 50.47(b)(10) and 10 CFR Part 50, Appendix E, Section IV.

Amended Emergency-Plan Change Process: The final rule ensures that (1) the effectiveness of the emergency plans will be maintained, (2) changes to the approved emergency plan will be properly evaluated, and (3) any change that reduces the effectiveness of the plan will be reviewed by the NRC before implementation. These requirements are incorporated in 10 CFR 50.54(q) and 10 CFR Part 50, Appendix E, Section IV.B.

Removal of Completed One-Time Requirements: The final rule eliminates several regulatory provisions that required holders of licenses to take certain one-time actions to improve the state of emergency preparedness following the Three Mile Island incident in 1979. These actions are complete and the requirements are no longer binding on any current licensee. The completed one-time requirements were removed from 10 CFR 50.54(r), 10 CFR 50.54(s)(1), 10 CFR 50.54(s)(2)(i), and 10 CFR 50.54(u).

Question No. 199

1. Please explain the validation process, if any, of Federal incident management plans coordinated by DHS.

2. Please explain whether the licensee's activities during the exercise are coordinated with Federal response in the National Incident Management System (NIMS) or not.

Answer:

1. Interagency documents coordinated by the U.S. Department of Homeland Security are vetted through a process outlined by Presidential Policy Directive-1, which prescribes the establishment of National Security Council Interagency Policy Committees to manage day-to-day policy and planning activities that are important to national security. When a new national plan is created or an existing national plan is revised, the appropriate policy committee reviews and provides feedback on the proposed plan. When a final revision is produced, those interagency partners concur on the plan. This ensures that all stakeholders are aware of their responsibility and commit to performing their role when called. Plans are also validated through their use in the National Exercise Program administered by FEMA.

2. Licensees are not currently required to adopt the National Incident Management System (NIMS); however, they respond to events using similar principles. Licensees typically coordinate directly with the Federal government through the NRC. The NRC response program has been developed to be compatible with the licensee response programs and with NIMS. This provides an effective interface between the licensees and Federal responders.

Question No. 224

Although the NRC will determine if policy changes are necessary with regards to pre-staging of KI [potassium iodide] beyond 10 miles, is any consideration being given to require pre-distribution to homes and businesses within 10 miles?

Answer: Each State that includes potassium iodide in its emergency plan is also responsible for the potassium iodide distribution plan to the population located within the 10-mile emergency planning zone. Some States with large transient populations decide to stockpile potassium iodide for post-accident distribution at designated centers; other States with smaller populations have distributed potassium iodide through the postal system. The most common type of distribution plan includes a combination of predistribution to citizens and schools and stockpiles in reception centers or other designated centers for distribution in the post-accident phase.

ARTICLE 17. SITING

Each Contracting Party shall take the appropriate steps to ensure that appropriate procedures are established and implemented for

(i) evaluating all relevant site-related factors that are likely to affect the safety of a nuclear installation for its projected lifetime

(ii) evaluating the likely safety impact of a proposed nuclear installation on individuals, society, and the environment

(iii) re-evaluating, as necessary, all relevant factors referred to in subparagraphs (i) and (ii) so as to ensure the continued safety acceptability of the nuclear installation

(iv) consulting Contracting Parties in the vicinity of a proposed nuclear installation, insofar as they are likely to be affected by that installation and, on request, providing the necessary information to such Contracting Parties, in order to enable them to evaluate and make their own assessment of the likely safety impact on their own territory of the nuclear installation

This section explains the NRC's responsibilities for siting, which include site safety, environmental protection, and emergency preparedness. First, this section discusses the regulations applying to site safety and their implementation, emphasizing regulations applying to seismic, geological, hydrological, meteorological, and radiological assessments. Next, it explains environmental protection, reevaluation of site-related factors, and lessons learned from Fukushima. Article 16 of this report discusses emergency preparedness and international arrangements, which would apply to Contracting Parties in obligation (iv) above.

Question No. 148
The report states that siting facilities away from densely populated areas is a principal component of NRC's defense-in-depth safety philosophy. It is also said that the evaluation of population distributions and the creation of restricted-use zones around a proposed facility are essential elements of compliance with regulatory requirements in 10 CFR Part 100 (Article 17; Section 17.2.2; page 118). Could you provide information about if the population's potential growth is regarded, for example, in monitoring programs considering that although not directly affect the exclusion zone, the population growth could lead to changes in use of water and soil or even modify access roads to the plant.
<u>Answer</u>: The population distribution within 50 miles of the proposed installation is addressed in the license application as a part of evaluating (1) all relevant site-related factors that are likely to affect the safety of a nuclear installation for its projected lifetime and (2) the likely safety impact of the proposed nuclear installation on individuals, society, and the environment. This includes the current population distribution and also potential population growth. Population growth is analyzed by making population-distribution projections for the area within 50 miles of the site, as well as population-density projections for the area within 20 miles of the site for the life of plant operation. This information is used in the evaluations of plant safety and of environmental impacts such as those from water use, land use, and socioeconomic factors over

the operational lifetime of the proposed installation. No residents are permitted within the exclusion-area boundary, which is controlled by the licensee. The low-population zone is selected so that appropriate protective measures could be taken in the event of a serious emergency. Growth in the low-population zone is expected of be very minimal. Population growth in areas outside the exclusion-area boundary and low-population zone could lead to some changes. However, these changes are evaluated in light of the projected growth, determined as discussed above, covering the operational life of the proposed plant. For emergency planning, periodic exercises and assessments are performed to evaluate the adequacy of emergency plans and institute alternative approaches, if required.
Question No. 149
Regarding the "low population zone" mentioned as part of the creation of restricted-use zones around a proposed facility. Is there any regulatory restriction for the establishment of industries or new inhabitants? (Article 17; Section 17.2; pages 188 / 189)
Answer: No residents are permitted within the exclusion-area boundary, which is controlled by the licensee. The low-population zone is selected so that the appropriate protective measures could be taken in the event of a serious emergency, so population growth in the zone is expected to be very minimal. Although there is no regulatory restriction against the establishment of industries or the addition of new inhabitants in the low-population zone, the licensee and the NRC maintain awareness of any significant changes in site environment in order to assess and institute any mitigation measures or engineering safeguards, if required.
Question No. 150
It is mentioned that several technical concerns were identified in the analyses supporting the regulation addressing the environmental impacts of spent nuclear fuel storage after the licensed lifetime of reactor operations (Article 17; Section 17.3.2; page 193). These technical concerns included the possible stored spent fuel deterioration? Could you give more details about the related actions foreseen to face these issues?
Answer: The technical concerns discussed in the cited CNS report text refer to deficiencies identified by the U.S. Court of Appeals related to the environmental analyses performed to support the NRC's generic determinations regarding the safety and environmental impacts of post-licensed life storage of spent fuel. The NRC has initiated an effort to reassess the environmental impacts of post-licensed life storage, including consideration of the technical concerns identified by the Court. Refer to the *Federal Register* notice at 78 FR 56776 titled "Waste Confidence—Continued Storage of Spent Nuclear Fuel," and dated September 13, 2013 (ADAMS Accession No. ML13256A004). With respect to the issue of potential deterioration of spent fuel that might be stored for a long time, the NRC is a participant in the EPRI's Extended Storage Collaboration Program. The program was established to investigate aging effects and mitigation options for the extended storage and transportation of used nuclear fuel and high-level waste. (See EPRI Report No. 1022914, "Extended Storage Collaboration Program (ESCP)—Progress Report and Review of Gap Analyses," August 2011, http://www.epri.com/abstracts/Pages/ProductAbstract.aspx?ProductId=000000000001022914). In addition to its participation in the Extended Storage Collaboration Program, the NRC is examining the technical needs of and potential changes to the regulatory framework that might be needed to continue licensing of spent fuel storage over longer periods. As part of this effort, the NRC staff prepared and issued a draft report presenting the results of its evaluation of the technical information needs for continued extended dry storage, and for the subsequent transportation of spent nuclear fuel following long-term storage. (See "Draft Report for Comment—Identification and Prioritization of the Technical Information Needs Affecting

Potential Regulation of Extended Storage and Transportation of Spent Nuclear Fuel," May 2012, Accession No. ML120580143). To date, deterioration of spent fuel has not been identified as a significant issue during license-renewal reviews of five independent spent fuel storage installations. (See DOE/EM-0654, Revision 3, "United States of America Fourth National Report for the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management," September 2011, http://energy.gov/sites/prod/files/4th_US%20_Nat%20_Report%20%2009-21-11%20%282%29.pdf.) The NRC continues to review applicants' AMPs to ensure that potential concerns with spent fuel deterioration are identified and mitigated before there is an adverse effect on safety.

Question No. 151

As heat sink unavailability is a safety related event, does the USA request assessment about water supply availability if drought causes low water levels?

Answer: Yes, the NRC requires that all safety-related ultimate heat-sink SSCs be designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions (Appendix A to 10 CFR Part 50). Subsection III(1), "Low Water from Drought," of Section 2.4.11, "Low Water Considerations," of NUREG-0800 (the Standard Review Plan) provides review procedures for consideration of low water at the ultimate heat sink. The Standard Review Plan states that "In cases where a common source of cooling water for operation and safety is provided, and where operation can affect minimum levels required for safety, the system will be acceptable if technical specifications are provided for shutdown before the ultimate heat sink can be adversely affected." RG 1.27, "Ultimate Heat Sink for Nuclear Power Plants," states that "because of the importance of the UHS to safety, these functions should be ensured during and following the design bases events postulated for the site (e.g., the Safe Shutdown Earthquake, design basis tornado, hurricane, flood, or drought)."

Question No. 152

Could the USA present the results of the reassessments of seismic and flooding hazards in response to Fukushima Daiichi NPP event?

Answer: At this time the seismic and flooding reevaluations are ongoing. On completion of the reassessments, the NRC is willing to present key results.

For current information on seismic reevaluations, please visit http://www.nrc.gov/reactors/operating/ops-experience/japan-dashboard/seismic-reevaluations.html.

For current information on flooding reevaluations, please visit http://www.nrc.gov/reactors/operating/ops-experience/japan-dashboard/flooding.html.

Question No. 153

Can you give more information about the regulatory guidance and the methodologies applied in the re-evaluation of the hazards due to earthquakes and flooding? (Article 17; Section 17.4; page 206)

Answer: The regulatory guidance and methodologies mentioned in Article 17, Section 17.4, page 194, are discussed more fully in Article 18, Section 18.5, pages 206 through 208. However, you can find full versions of the guidance and methodology documents on the NRC Web site as follows:

For current information on seismic reevaluations, please visit http://www.nrc.gov/reactors/operating/ops-experience/japan-dashboard/seismic-reevaluations.html.

For current information on flooding reevaluations, please visit http://www.nrc.gov/reactors/operating/ops-experience/japan-dashboard/flooding.html.

Question No. 200

Are there any NPP monitoring systems operated by either the Utility or the Regulatory body for site characteristics, such as earthquakes, surface faulting, groundwater elevation, slope failures, foundation subsidence etc., and are these information open to the public, if any?

Answer: The NRC does not operate any monitoring systems as part of its regulatory activities. If required, as part of their licensing requirements, the licensees may establish monitoring systems and report their findings to the NRC if and when there is an unusual event and/or activity. Currently, none of the operating nuclear power plants in the United States runs a seismic network to monitor local and regional seismic activity. However, the operating nuclear power plants do maintain strong-motion accelerographs to monitor ground motions at the foundation level and/or within structure. These accelerographs serve the purpose of monitoring and informing the plant operators and the regulators about the ground-motion levels should there be an earthquake nearby affecting the plant. This information might trigger a series of plant actions, including immediate plant shutdown, if necessary.

Surface faulting does not require continuous monitoring within the U.S. regulatory framework, because extensive geological analyses are conducted before the licensing and construction of nuclear power plants. No plant has been built over a known active fault in the United States. Should new information and data indicate that a fault previously identified as inactive might be an active fault, U.S. regulations require that adequate protective measures be taken, including, if necessary, to shut down the nuclear power plant.

Similar to the strong-motion sensors, some nuclear power plants have requirements in their license to operate water-level and temperature gauges at intake structures, outlet works, and/or other holding structures at or near the site. The water-level gauges alert site personnel when the water level is above or below a preset range, while the temperature gauges typically alert personnel when the temperature is above a predetermined point. Depending on the nuclear power plant's license requirements, these abnormal water levels and temperatures can trigger a series of plant actions, ranging from deployment of water protection features to potentially shutting down the plant.

Similarly to surface faulting, slope failure is also addressed extensively in the U.S. regulations during the siting phase. No additional monitoring activities are conducted.

Foundation subsidence or settlement of soils beneath the nuclear power plant during and after construction receives a great deal of attention in the U.S. regulatory framework. Those sites found to have potentially large amounts of settlement and foundation subsidence may be required to monitor the settlement activity at their sites continuously during the construction and operational phases of the nuclear power plants.

Any monitoring data is usually not available publicly in real-time; however, data and results might be publicly available in industry reports submitted to the NRC and/or in NRC documents and reports.

Question No. 225

The NRC request for licensees to re-evaluate site-specific seismic and flooding hazards may have resulted in physical changes or safety systems enhancements, but the report is not explicit in this area. Also, extending the scope of re-evaluation to cover other natural hazards, such as hurricanes and tornadoes, may have been considered but less visible in the report. Can the USA elaborate on these areas?

Answer: In response to the Fukushima accident, the NRC used its regulatory processes to request that licensees reevaluate the seismic and flooding hazards at their sites using present-day regulatory guidance and methodologies. The reevaluations are ongoing. Once the reevaluations are complete, the results will be used to determine whether additional regulatory actions are necessary to ensure plants are adequately protected from seismic and flooding events.

For current information on seismic reevaluations, please visit http://www.nrc.gov/reactors/operating/ops-experience/japan-dashboard/seismic-reevaluations.html.

For current information on flooding reevaluations, please visit http://www.nrc.gov/reactors/operating/ops-experience/japan-dashboard/flooding.html.

Seismic and flooding hazards reevaluations are Tier 1 Fukushima lessons-learned activities that have been addressed without unnecessary delay. Implementing lessons learned for other external hazards (e.g., tornadoes, hurricanes, and drought) were Tier 2 Fukushima lessons-learned activities that could not be initiated because of a need for further technical assessment and alignment, dependence on Tier 1 issues, or a lack of availability of critical skill sets. These issues subsequently were included with the NTTF Recommendation 2.2 Program Plan for Periodic Confirmation of Seismic and Flooding Hazards. Discussion of the status of these activities can be found in SECY 12-0095 (http://www.nrc.gov/reading-rm/doc-collections/commission/secys/2012/2012-0095scy.pdf) and SECY 13-0095 (http://www.nrc.gov/reading-rm/doc-collections/commission/secys/2013/2013-0095scy.pdf).

Question No. 226

Does the NRC impose a reporting obligation on the facilities it has licensed to monitor and confirm that the original site approval conditions remain consistent with the current understanding of external hazards? Can you elaborate on how the NRC monitors for changes to the hazards basis or other factors that were originally used to determine the safety of a site, for example that the area remains an area of low population?

Answer: Reporting obligations depend on whether the licensee identifies new information that adversely affects its final safety analysis report (FSAR). If the licensee identifies such information, the FSAR requires updating in accordance with 10 CFR 50.71(e). This is separate from reporting of an event or condition under 10 CFR 50.72 and 50.73, as well as the requirements under 10 CFR 50.54(ff) and Paragraph V.(a)(ii) of Appendix A, “Seismic and Geologic Siting Criteria for Nuclear Power Plants,” to 10 CFR 100. In addition, 10 CFR 50.9, “Completeness and Accuracy of Information,” requires licensees to “notify the Commission of information identified by the applicant or licensee as having for the regulated activity a significant implication for public health and safety or common defense and security.”

Further, and separate from the reporting requirements just described, the NRC has an ongoing generic-issues program which provides another potential mechanism to assess issues related to changing external hazard conditions at a site. In order for an issue or concern to be admissible into the program, a number of criteria must be met as stated in NRC Management Directive 6.4, “Generic Issues Program.”

The NRC differentiates external hazards into two categories, man-made and natural. But the specific issue of changes in population density would appear to be more related to emergency-planning issues than generic issues. To date, changes in plant site conditions related to man-made impacts (e.g., from pipelines, marine terminals, air-traffic patterns, highway use, and industrial facilities) have not met the threshold for inclusion in the generic-issues program because existing regulatory programs, processes, and guidance provide adequate mechanisms to address this issue. For example, industry document NEI 98-03, "Guidelines for Updating Final Safety Analysis Reports," states that … "licensees should evaluate potentially significant changes in the site environs (e.g., a new natural gas line within the site boundary or a major new industrial facility near the plant site) to determine whether notification of NRC and appropriate update of the UFSAR are required."

Natural external events, such as the impact on plant safety resulting from (for example) new, potentially more severe seismic loads or previously unanalyzed flooding events, have been included in the generic-issues program. Generic Issues 199 and 204, respectively, addressed seismic and flooding issues and were identified before the accident in Japan. However, as a result of that accident, they have now been folded into agency initiatives addressing Fukushima-related plant enhancements.

ARTICLE 18. DESIGN AND CONSTRUCTION

Each Contracting Party shall take the appropriate steps to ensure that:

(i) **the design and construction of a nuclear installation provides for several reliable levels and methods of protection (defense in depth) against the release of radioactive materials, with a view to preventing the occurrence of accidents and to mitigating their radiological consequences should they occur**

(ii) **the technologies incorporated in the design and construction of a nuclear installation are proven by experience or qualified by testing or analysis**

(iii) **the design of a nuclear installation allows for reliable, stable, and easily manageable operation, with specific consideration of human factors and the man-machine interface**

This section explains the defense-in-depth philosophy and how it is embodied in the general design criteria of U.S. regulations. It explains how applicants meet the defense-in-depth goals and how the NRC reviews applications and conducts inspections before issuing licenses to ensure that this philosophy is implemented in practice. Next, this section discusses measures for ensuring that the applications of technologies are proven by experience or qualified by testing or analysis. Finally, this section discusses requirements for reliable, stable, and easily manageable operation (specifically considering human factors and the man/machine interface) and lessons learned from Fukushima. Article 12 of this report also provided information on the human-factors obligations.

Question No. 154
Art. 18 Section 18.1.1, 2nd par, p. 197: "The NRC staff amplified its defense-in-depth philosophy in RG 1.174, which provides guidance on using a PRA in risk-informed decisions on plant-specific changes." Could you please explain how could the probabilistic risk assessment (PRA) be used to amplify fulfillment of the defence in depth, considering that this concept is fundamentally deterministic? Reasoning: The referenced RG 1.174 is suggesting that the PRA can "provide insights into whether the extent of defense-in-depth is appropriate", namely verify that "a reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation". Yet, the deterministic nature of the defence in depth seems to require that all these goals are met in the same time and to the extent reasonably achievable - the requirement to achieve a balance between these goals could justify and encourage a somewhat limited effort to achieve those of these goals that have been already somehow covered.
<u>Answer</u>: Defense in depth is achieved by providing multiple diverse layers of protection (that is, by preventing accidents, mitigating the effects of accidents, and practicing emergency preparedness), each capable of being effective should another fail. For example, reactor-vessel and containment barriers can be effective in protecting the public from radioactivity released from failed fuel during an accident, and emergency preparedness can be effective in protecting the public from radioactivity released from a failed reactor vessel and containment building in a severe accident. In the context of guidance provided in RG 1.174, a reasonable balance of these layers helps to ensure an apportionment of the plant's capabilities between limiting

disturbances to the plant and mitigating their consequences. "Balance" is not meant to imply an equal apportionment of capabilities. A reasonable balance is preserved if the proposed plant change does not significantly reduce the effectiveness of a layer that exists in the plant design before the proposed change. The NRC recognizes that aspects of a plant's design might cause one of the three layers to be adversely affected. For these situations, the balance between the other two layers becomes especially important when evaluating the impact of a proposed change to the licensing basis and its impact on defense in depth.

The NRC agrees that the defense-in-depth element is derived from traditional engineering considerations. It is not intended that the assessment of defense in depth needed to support a risk-informed change to the current licensing basis depend on risk insights arising from probabilistic risk-assessment models. However, a comprehensive risk analysis can provide quantitative and qualitative insights regarding the extent to which a proposed change might affect the balance among the levels of defense in depth (i.e., balance among preventing accidents, mitigating accidents, and emergency preparedness). Quantitative and qualitative risk insights can also be useful in identifying any adverse impact of a proposed change on the effectiveness of the fission-product barriers. Risk information can also help identify how a proposed change could affect programmatic activities; system redundancy, independence, and diversity; common-cause failure; and reliance on operator actions.

Question No. 155

Could the USA clarify, or give some references about, defense-in-depth level independence considerations? Does the USA have any requirements or criteria for defense-in-depth level independence assessment? Does the USA consider necessary General Design Criteria (10 CFR Part 50) to be updated, taking into account lessons learned after the Fukushima Daiichi NPP event?

Answer: A substantial reduction in the ability to accomplish a safety function is not consistent with the defense-in-depth philosophy. A safety function might be compromised (meaning that its system redundancy, independence, and diversity would not be preserved) when a proposed change would introduce new dependencies among plant equipment or would defeat one of the plant features that provides system redundancy, independence, or diversity. The introduction of new dependencies could reduce the level of redundancy, independence, or diversity for fulfilling a safety function. One form of dependency is the possibility of common-cause failure mechanisms. Reduction in system redundancy, independence, or diversity can potentially result in significant reduction in the effectiveness of one of the defense-in-depth layers that exists in the plant design before the proposed change or might reduce the effectiveness of one of the fission product barriers.

Currently, defense in depth is not an explicit requirement for plants regulated by the NRC except in the specific area of fire protection. Rather, it is a key element of the NRC safety philosophy that has guided the NRC, and the Atomic Energy Commission before it, and, as such, is embodied in the current regulatory requirements and implementing guidance.

The NRC currently has several ongoing activities that could affect the way defense in depth is applied in regulatory processes in the future:

- Commission decision on the NRC staff's recommendations (SECY-13-0132, http://www.nrc.gov/reading-rm/doc-collections/commission/secys/2013/2013-0132scy.pdf) for disposition of the NRC Fukushima NTTF's recommendation that the Commission establish a logical, systematic, and coherent regulatory framework for adequate protection that appropriately balances defense in depth and risk

- considerations (in “Recommendations for Enhancing Reactor Safety in the 21st Century,” dated July 12, 2011 (ADAMS Accession No. ML111861807)).
- NRC staff consideration of the nuclear power-reactor recommendations presented in the Risk Management Task Force Report, NUREG-2150, “A Proposed Risk Management Regulatory Framework,” dated April 2012 (ADAMS Accession No. ML12109A277).
- Revision of RG 1.174 to better address defense in depth; see proposed Revision 3 (Draft Guide 1285) issued in May 2012 (ADAMS Accession No. ML12012A006).

Question No. 156

Post Fukushima Lessons Learned: In June 2013, the NRC modified the Order that required reliable hardened vents for BWRs with Mark I and Mark II containments to require that those vents remain functional in the conditions following reactor core damage.
Please provide further clarification why only for the Mark I and Mark II containments hardened vents are required? Will these venting systems be equipped with an aerosol filter?

Answer: All U.S. containment designs are encompassed in the lessons learned in light of the nuclear accident at Fukushima. A reliable hardened vent for Mark I and Mark II containments (Recommendation 5.1) is a Tier 1 recommendation for which actions are to begin without unnecessary delay. A reliable hardened vent for other containment designs (Recommendation 5.2) is a Tier 3 recommendation that requires longer-term evaluation. The NRC will use insights from the development and implementation of requirements for the Mark I and Mark II containments to inform its evaluations of what, if any, changes should be proposed for other containment designs.

Licensees of nuclear power plants with BWR Mark I and Mark II containments must install hardened vents able to function following a severe accident. In addition, the NRC is evaluating possible regulations that would require each licensee to develop a filtering strategy and improve accident-management capabilities for maintaining containment integrity. Engineered filtering systems are one of the options being evaluated as part of the rulemaking process.

For more information, please visit http://www.nrc.gov/reactors/operating/ops-experience/japan-dashboard/hardened-vents.html.

Question No. 157

The NRC also began a SFP (Spent Fuel Pool) study, which considered a SFP similar to the one at Fukushima and 23 U.S. reactors, and an earthquake several times stronger than what the SFP’s design considered. The study examined both a full SFP and one with less fuel and more spacing between individual fuel assemblies, as well as emergency procedures for adding water to the pool in the unlikely event that the earthquake causes the pool to lose water. The detailed analysis showed that even a very strong earthquake has a low probability of damaging the pool to the point of losing water. The draft study also showed that even if this particular pool was damaged, the fuel could be kept cool in all but a few exceptional circumstances.

Which additional mitigating measures (e.g. mobile pumps for adding water) were considered in this draft study?

Answer: The treatment of mitigation is described in detail in Section 5.3 of the spent fuel pool study (ADAMS Accession No. ML13256A334 or http://pbadupws.nrc.gov/docs/ML1325/ML13256A334.html). In that study, the installed accident-mitigation equipment was assumed to be damaged by the event and no credit was given to recovery or repair. The study assumed that, for the mitigated scenarios, actions associated with the regulatory requirement of 10 CFR 50.54(hh)(2) (http://www.nrc.gov/reading-rm/doc-collections/cfr/part050/part050-0054.html) are successfully performed and additional onsite and offsite resources are available to extend the utility of this equipment until onsite capabilities can be recovered. The 10 CFR 50.54(hh)(2) equipment consists of additional power sources and portable pumps. For the actual mode of operation, the study assumed either a 500-gpm makeup or 200-gpm spray based on the requirements in NEI 06-12, "B.5.b Phase 2 & 3 Submittal Guideline."

Question No. 158

On page 256 it is stated: Most U.S. plants have 4-hour coping durations for mitigating SBO conditions. U.S.-plants also developed emergency response strategies to mitigate the effects of fires postulated to adversely affect safety system functions. In many cases, stations rely on SBO diesel generators, gas turbines, or ac power from other onsite sources to mitigate the blackout condition…

On page 257: The following four specific recommendations were provided to the U.S. industry:
1. For all units, develop methods to maintain (or restore) core cooling, containment integrity, and SFP inventory using existing installed and portable equipment during an extended loss of electrical ac power event that lasts at least 24 hours.

In view of the generic nature of this issue, please provide more details of the evaluations and findings achieved so far. Is an increase of battery capacity also considered as an option for mitigating beyond SBO conditions?

Answer: All U.S. plants have conducted analyses and developed strategies for using permanently installed and onsite portable equipment during an extended loss of AC power that lasts at least 24 hours. The approach used is described in NEI 12-06, which is available on the NRC's Web site at http://pbadupws.nrc.gov/docs/ML1224/ML12242A378.pdf. Initially, installed equipment will be used. For plants with limited battery capacity, adding batteries might have been considered, but the typical approach has been to develop strategies for early load shedding to extend operability of equipment long enough for portable equipment, including battery chargers, to be put into place for longer-term operation.

Question No. 159

Is all plant modernizations also monitored in the described manner? If not all; who decides what modernizations should be monitored (NRC, NPP) and according to what criteria?

Answer: The NRC has a number of criteria, regulations, processes, and audit and inspection procedures associated with plant modernizations and their monitoring. In accordance with the license-renewal regulation (10 CFR Part 54), the NRC performs a safety review of the applicant's license-renewal application to determine whether the applicant has adequately demonstrated that the effects of aging will not adversely affect any SSCs, as identified in 10 CFR 54.4. The applicant must demonstrate that the effects of aging will be managed in such a way that the intended functions of the passive and long-lived SSCs (e.g., the reactor vessel and reactor coolant system boundary) will be maintained during the period of extended operation because these SSCs are not subject to replacement based on a qualified life or specified time period (10CFR 54.21(a)(1)(i) and (ii)). For active components (e.g., motors, diesel generators, relays), surveillance and maintenance programs will continue throughout the period of extended operation.

Managing the aging of these SSCs implies ensuring the availability of their required safety functions throughout the life cycle of the plant, including changes and modifications that occur over time and expected wear and tear. This requires addressing both the physical aging of SSCs (which results in degradation of their performance characteristics) and the need for modernization of SSCs when they become obsolete with respect to the current state of knowledge, standards, and/or technology.

The GALL Report focuses on management of physical aging. Modernization of SSCs important to safety because the SSCs have become obsolete is managed proactively throughout their service life. Aspects of technological obsolescence, such as insights into individual degradation mechanisms, have already been taken into consideration in the GALL AMPs. New insights will be addressed in future AMP updates.

In addition, 10 CFR 50.59 establishes the applicable regulatory requirements informing licensees as to which changes can and cannot be made without explicit NRC licensing review and approval. For changes that do not require NRC review, the licensee must maintain a record of the changes. The record must contain a written evaluation which provides the bases for the determination that the changes do not require a license amendment. Licensees must submit a report containing a brief description of any changes, including a summary of the evaluation, every 24 months.

Recognizing the importance of design control to safety cornerstones that cannot be fully monitored through performance indicators, the ROP samples plant modifications to ensure that the licensee is managing design changes in a way that is safe and compliant with regulatory requirements. This is currently accomplished in accordance with IP 71111.17T, "Evaluations of Changes, Tests and Experiments and Permanent Plant Modifications," and IP 71111.18, "Plant Modifications."

Question No. 160

Does the NRC vendor audit for safety related equipment substitutes the licensees own supplier audit or are they performed in parallel? If parallel; how is the communication/experience exchange between the two groups handled?

Answer: The NRC's independent inspections of vendors are separate from the licensees' supplier audits that are required under Appendix B to 10 CFR 50. Licensees are required to perform audits of their suppliers regardless of whether the NRC performs an inspection of the same supplier or not.

Question No. 161

Are all new applications expected to get the same attention as Watts Bar 2 (dedicated teams at both head quarters)?

Answer: The AP1000® plants being constructed under 10 CFR Part 52 combined licenses are the first of their kind and construction has been from the ground up. The construction of those plants has received (and will continue to receive) greater attention than the construction at Watts Bar 2. The Watts Bar 2 construction project was partially complete when construction was stopped in the mid-1980s; there has been less construction work to inspect following the restart. In addition, the Watts Bar 2 design and construction is similar to that for Watts Bar 1; thus, the NRC has more experience with the design. Inspectors at the NRC Region II office inspecting the Watts Bar 2 construction are the same personnel inspecting the AP1000 construction sites. At NRC headquarters, the Watts Bar 2 project is managed by NRR. The AP1000 construction project is managed by NRO.

Question No. 162

It is specified that NRC carries out it's own Vendor inspection programme. What is the legal basis for such inspections and how does the Licensee interact in the process?

Answer: 10 CFR 21, “Reporting of Defects and Noncompliance,” which is derived from Section 206 of the Energy Reorganization Act of 1974, is directly applicable to any individual director or responsible officer of a firm constructing, owning, operating, or supplying the components of any facility or activity which is licensed or otherwise regulated under the AEA or the Energy Reorganization Act of 1974. In addition, licensees pass down the requirements of 10 CFR 50, Appendix B, through procurement contracts which allow the NRC to ensure that those requirements are being met. Note that any findings relating to 10 CFR 21 are written as Notices of Violation because they are directly applicable to the vendor and any findings relating to 10 CFR 50, Appendix B, are written as Notices of Nonconformances because the vendor did not conform to the procurement contract’s requirements. All inspection reports are made public and available to the licensees to assess whether any findings are applicable to them.
Question No. 163
The NRC has also implemented a significant and continuing research program in cyber security for digital plant control systems. Please describe in more detail the highlights of this research program.
Answer: Details on the Cyber Security research program can be found in Section 3.2, “Security Aspects of Digital Systems,” of the “NRC Digital System Research Plan FY 2010-FY 2014” (ADAMS Accession No. ML100541484) under research topics (subsections) 3.2.1, “Security of Digital Platforms,” and 3.2.2, “Network Security.” Under these research topics, the NRC conducted cyber vulnerability studies of several digital platforms approved for use in U.S. nuclear power plants and investigated network cyber vulnerabilities. Additionally, the NRC collaborates with other U.S. Federal agencies’ cyber research programs.
Question No. 164
What happens when during construction important lessons are learned from (possibly other) construction projects or operating experience? Does the licensee have the obligation to follow up on these and to improve the design?
Answer: The NRC collects, screens, evaluates, and communicates lessons learned from operating and construction experience, including applicable international experience, as available. Depending on the significance of a lesson learned, the NRC may develop a generic communication publication, issue a proposal for new regulations, conduct additional study, or revise inspection procedures. A licensee’s obligation to follow up on construction lessons depends on the results of the NRC’s evaluation, which considers both qualitative and quantitative factors of significance. For the most significant events, the NRC might issue an Order, which requires action by a licensee. NRC Orders require written responses and the completion of actions within specified time frames. In other cases, the NRC could publish a Bulletin, which requests licensee actions and/or information to address significant issues regarding matters of safety, security, or environmental significance that have great urgency. NRC Bulletins require a written response. For some situations, the NRC might publish an IN, which communicates operating or analytical experience to the nuclear industry. The industry is expected to review the information and consider appropriate actions to avoid similar problems, but INs do not obligate licensees to take action.
Question No. 165
Could the USA specify which safety objectives are associated with a post-Fukushima reassessment (reduction of core damage frequency, reduction of releases, etc.)? Could the USA clarify if prioritization is given to seismic and flooding hazards, whereas earthquake or flooding may not cause the main risks due to external hazards for every site?

Answer: In the United States, a risk-informed process considers core-damage frequency and large early-release frequency to establish regulatory priorities.

Seismic and flooding hazards reevaluations are Tier 1 Fukushima lessons-learned activities that have been addressed without unnecessary delay. Implementing lessons learned for other external hazards (e.g., tornadoes, hurricanes, and drought) are Tier 2 Fukushima lessons learned activities that could not be initiated immediately because of a need for further technical assessment and alignment, dependence on Tier 1 issues, or a lack of availability of critical skill sets. These issues subsequently were included with the NTTF Recommendation 2.2 Program Plan for Periodic Confirmation of Seismic and Flooding Hazards. Discussion of the status of these activities can be found in SECY-12-0095 (http://www.nrc.gov/reading-rm/doc-collections/commission/secys/2012/2012-0095scy.pdf) and SECY-13-0095 (http://www.nrc.gov/reading-rm/doc-collections/commission/secys/2013/2013-0095scy.pdf).

Question No. 166

18.5.1 Seismic, Flooding and Other Hazards Protection
Regarding the evaluation of site specific seismic hazards, please clarify if it includes reevaluation of methodology for establishing design basis earthquake included in licensing bases, or just reevaluation of recent seismic data for reassessing adequacy of existing seismic margins.

Answer: On December 23, 2011, the Consolidated Appropriations Act, Public Law 112-074, was signed into law. Section 402 of the law requires a reevaluation of licensees' design basis for external hazards and expands the scope to include other external events, as described below:

"The Nuclear Regulatory Commission shall require reactor licensees to re-evaluate the seismic, tsunami, flooding, and other external hazards at their sites against current applicable Commission requirements and guidance for such licensees as expeditiously as possible, and thereafter when appropriate, as determined by the Commission, and require each licensee to respond to the Commission that the design basis for each reactor meets the requirements of its license, current applicable Commission requirements and guidance for such license. Based on the evaluations conducted under this section and other information it deems relevant, the Commission shall require licensees to update the design basis for each reactor, if necessary."

On March 4, 2012, NRC staff issued to reactor licensees a request for information under the requirements of 10 CFR 50.54(f) requesting, in addition to other Fukushima-related lessons-learned information, plant-specific seismic reevaluations in accordance with NTTF Recommendation 2.1 and consistent with the Consolidated Appropriations Act.

In November 2012, the U.S. nuclear industry submitted EPRI 1025287, "Seismic Evaluation Guidance." This guidance outlines a process and provides guidance for investigating the significance of new estimates of seismic hazard and, where necessary, performing further seismic evaluations. This guidance is primarily designed for use in responding to the U.S. NRC's NTTF Recommendation 2.1, "Seismic." The guidance includes a screening process for evaluating updated site-specific seismic-hazard and ground-motion response-spectrum estimates against the plant's safe-shutdown earthquake and "high confidence of low probability of failure" capacities. It also provides selected seismic risk-evaluation criteria as well as spent fuel pool evaluation criteria.

The NRC endorsed EPRI 1025287 on February 15, 2013.

In April 2013, the U.S. nuclear industry submitted EPRI 3002000704, “Augmented Approach for the Resolution of Fukushima NTTF Recommendation 2.1: Seismic,” which describes an expedited evaluation process that addresses interim evaluations of critical plant equipment to be implemented before performing the complete plant seismic risk evaluations described in EPRI 1025287.

The NRC endorsed EPRI 3002000704 on May 7, 2013.

For more information on seismic reevaluations, please visit http://www.nrc.gov/reactors/operating/ops-experience/japan-dashboard/seismic-reevaluations.html.

Question No. 167

18.5.1 Seismic, Flooding and Other Hazards Protection
In relation with the evaluation of other external hazards, could you please give more details about prioritization and schedule of analysis to be performed by licensees and possible regulatory actions by NRC?

Answer: In the United States, a risk-informed process that considers core-damage frequency and large early-release frequency is used to establish regulatory priorities.

Seismic and flooding hazards reevaluations are Tier 1 Fukushima lessons-learned activities that have been addressed without unnecessary delay. Implementing lessons learned for other external hazards (e.g., tornadoes, hurricanes, and drought) were Tier 2 Fukushima lessons-learned activities that could not be initiated immediately because of a need for further technical assessment and alignment, dependence on Tier 1 issues, or a lack of availability of critical skill sets. These issues subsequently were included with the NTTF Recommendation 2.2 program plan for periodic confirmation of seismic and flooding hazards. Discussion of the status of these activities can be found in SECY-12-0095 (http://www.nrc.gov/reading-rm/doc-collections/commission/secys/2012/2012-0095scy.pdf) and SECY-13-0095 (http://www.nrc.gov/reading-rm/doc-collections/commission/secys/2013/2013-0095scy.pdf).

For detailed information on seismic reevaluations, please review documentation located at this link: http://www.nrc.gov/reactors/operating/ops-experience/japan-dashboard/seismic-reevaluations.html.

For detailed information or flooding reevaluations, please review documentation located at this link: http://www.nrc.gov/reactors/operating/ops-experience/japan-dashboard/flooding.html.

Question No. 168

Could you please provide a short overview of how NRC have considered Digital I&C in PRA and Deterministic safety analysis?

Answer: The deterministic safety analysis is considered to be the plant’s accident analysis demonstrating acceptable plant response in the event of an anticipated operational occurrence or postulated accident that is within the plant’s design basis. It also includes the plant analysis (thermo-hydraulic and nuclear) for beyond-design-basis events.

Within the plant’s PRA, quantitative values are provided for the digital instrumentation and control (I&C) system’s reliability, availability, etc., to provide support for the overall plant PRA. For the I&C safety review, the quantitative PRA values are not used to determine whether a digital instrument or control is sufficiently safe. The I&C safety review applies deterministic safety methods such as single-failure analysis, defense-in-depth and diversity analysis, and evaluation of the quality software development process. The reason why quantitative PRA values are not currently used in the safety decision for digital I&C systems is the lack of maturity in the area of digital I&C reliability assessment (particularly for the software aspects). Sensitivity analyses are conducted with the digital I&C quantitative PRA values to ensure that changes in those values do not have a significant impact on the overall plant PRA.

The plant's accident analysis provides a basis for the safety functions performed by the digital I&C system, including the setpoints and response times of the instruments and controls. I&C safety reviewers evaluate the information in the plant safety analysis and the design to ensure that the safety functions can be implemented given various operating conditions and failures. In cases in which a common-cause failure might disable redundant portions of the safety system (e.g., software common-cause failure within a digital safety I&C system), the NRC considers these failures to be beyond-design-basis events. However, applicants are required to address these failures by performing a defense-in-depth and diversity analysis. Part of the analysis is a plant accident analysis that uses realistic, nominal plant conditions and includes the common-cause failure of redundant digital safety I&C systems. The analysis demonstrates that in the event of digital I&C common-cause failures and an anticipated operational occurrence or postulated accident, the plant can respond and prevent radioactive material release or ensure that any release is below prescribed limits.

Question No. 169

Regarding reevaluation of hazards resulting from earthquakes, floods, and extreme weather conditions, a return period of 10.000 years is recommended by ENSREG/WENRA [Western European Nuclear Regulators Association] as a basis for the assessment of the adequacy of plant specific protection measures. In Switzerland, this number is embodied in the ordinance regarding threat assumptions and protection of NPPs (SR732.112.2). Could you please elaborate on the U.S. requirements regarding assessment of natural hazards?

Answer: For detailed information on seismic reevaluations, please review documentation located at this link: http://www.nrc.gov/reactors/operating/ops-experience/japan-dashboard/seismic-reevaluations.html

For detailed information or flooding reevaluations, please review documentation located at this link: http://www.nrc.gov/reactors/operating/ops-experience/japan-dashboard/flooding.html

Seismic and flooding hazards reevaluations are Tier 1 Fukushima lessons-learned activities that have been addressed without unnecessary delay. Implementing lessons learned for other external hazards (e.g., tornadoes, hurricanes, and drought) were Tier 2 Fukushima lessons-learned activities that could not be initiated immediately because of a need for further technical assessment and alignment, dependence on Tier 1 issues, or a lack of availability of critical skill sets. These issues subsequently were included with the NTTF Recommendation 2.2 Program Plan for Periodic Confirmation of Seismic and Flooding Hazards. Discussion of the status of these activities can be found in SECY-12-0095 (http://www.nrc.gov/reading-rm/doc-collections/commission/secys/2012/2012-0095scy.pdf) and SECY-13-0095 (http://www.nrc.gov/reading-rm/doc-collections/commission/secys/2013/2013-0095scy.pdf).

Question No. 191

Section 18.4 describes the new reactor construction experience program. Please provide additional information about the implementation of the lessons learned into the construction inspection activities. How has the program influenced factors like the frequency of inspections, the focus of inspections and procedures related to construction inspection activities?

Answer: The implementation (prompted by significant lessons learned) of changes to construction inspection activities is a key component of the NRC's operating and construction experience program. To date, the frequency of inspections has not been influenced by lessons learned, but the NRC will continue to assess the need for increased inspections as construction progresses. The focus of inspections has been influenced by the operating and construction experience program, which develops reports of notable lessons learned as part of semiannual assessments for all U.S. nuclear plants, including those under construction. The NRC refers to these assessments as mid-cycle and end-of-cycle reviews and uses them to inform future inspection plans. As an example, lessons learned from concrete degradation problems at the Seabrook nuclear plant have been used to focus some inspections on construction activities that could contribute to alkali-silica reaction. Insights from the operating and construction experience program are also incorporated in NRC inspection procedures by listing pertinent generic communications as reference documents and by providing specific inspection guidance resulting from lessons learned. For example, NRC IP 73758, "Part 52, Functional Design and Qualification, and Preservice and Inservice Testing Programs for Pumps, Valves and Dynamic Restraints" (http://pbadupws.nrc.gov/docs/ML1231/ML12314A205.pdf), includes Operating Experience in Section 03.09 of Attachment 3 and provides several generic communications in the References Section. NRC inspectors use this information for both planning and conducting inspections.
Question No. 192
Section 18.5 describes the NRC review and implementation of the NTTF recommendations. The report describes the efforts at re-evaluation of seismic and flooding hazard at NPPs in the US. How are the NRC programs taking into consideration the uncertainties in seismic and tsunami hazard prediction methods revealed by the event in Japan?
Answer: Uncertainties in the seismic and tsunami prediction methods are discussed in the guidance for conducting seismic and flooding reevaluations. For detailed information on seismic reevaluations, please review documentation located at this link: http://www.nrc.gov/reactors/operating/ops-experience/japan-dashboard/seismic-reevaluations.html. For detailed information or flooding reevaluations, please review documentation located at this link: http://www.nrc.gov/reactors/operating/ops-experience/japan-dashboard/flooding.html.
Question No. 227
The report identifies that some policy statements and regulatory documents were formulated and issued a few decades ago. The report is not explicit on discussing the impact of Fukushima follow-up on present day revisions to policy statements and regulatory documents. Can you elaborate further on this?
Answer: The regulatory documents listed in section 18.5 were provided as background to demonstrate the current state of U.S. design and construction regulation. In light of the accident at Fukushima, the Commission chartered the NTTF to conduct a systematic and methodical review of NRC processes and regulations to determine whether the agency should make additional improvements to its regulatory system. The NTTF, NRC staff, ACRS, and other stakeholders identified issues and made recommendations to the Commission regarding possible changes to agency requirements and policies. In October 2011, the Commission chartered the JLD under the direction of a senior staff steering committee to implement the NRC's longer-term review, including those items identified in the Chairman's tasking memorandum for longer-term review, recommendations for evaluation

that were provided by the Task Force and approved by the Commission, and any other review topics as directed by the Commission. The scope of the steering committee's review includes power and non-power reactors, non-operating reactors, and non-reactor NRC licensees and is informed by interactions with external stakeholders.

As part of this process, the United States is addressing the lessons learned in light of the accident at Fukushima and making appropriate revisions to regulatory requirements and updates to Commission policy and guidance documents.

Question No. 236

In Safety Standards No. SSR-2/1, IAEA described that Design Extension Conditions (DEC) shall be considered for the design of a new NPP. Does NRC have a plan to reflect DEC into regulation? If so, please explain the plan.

Answer: The NRC has processes and decision criteria for establishing new power-reactor design requirements on the basis of engineering judgment, operating experience, deterministic assessments, and probabilistic assessments, their purpose being the further improvement of safety of the nuclear power plant by enhancing the plant's capabilities to withstand, without unacceptable radiological consequences, accidents that either are more severe than design-basis accidents or involve additional failures. The NRC has done this for the following conditions: station blackout (10 CFR 50.63, "Loss of All Alternating Current Power"), anticipated transients without scram (10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transients without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants"), aircraft impact (10 CFR 50.150, "Aircraft Impact Assessment"), and loss of large areas of the plant because of explosion and fire (10 CFR 50.54(hh)(2)). The NRC is currently considering a recommendation (put forward in SECY-13-0132, "U.S. Nuclear Regulatory Commission Staff Recommendation for the Disposition of Recommendation 1 of the Near-Term Task Force Report") to formally recognize these requirements as design extension requirements and improve the process for developing such requirements in the future. In addition, the NRC staff is currently evaluating recommendations in the Risk Management Task Force Report, NUREG-2150 (ADAMS Accession No. ML12109A277). The result of this evaluation might lead to further development of a more logical and systematic framework for developing new design requirements.

ARTICLE 19. OPERATION

Each Contracting Party shall take appropriate steps to ensure that:

(i) **the initial authorization to operate a nuclear installation is based on an appropriate safety analysis and a commissioning program demonstrating that the installation, as constructed, is consistent with design and safety requirements**

(ii) **operational limits and conditions derived from the safety analysis, test, and operational experience are defined and revised as necessary for identifying safe boundaries for operation**

(iii) **operation, maintenance, inspection, and testing of a nuclear installation are conducted in accordance with approved procedures**

(iv) **procedures are established for responding to anticipated operational occurrences and to accidents**

(v) **necessary engineering and technical support in all safety related fields is available throughout the lifetime of a nuclear installation**

(vi) **incidents significant to safety are reported in a timely manner by the holder of the relevant license to the regulatory body**

(vii) **programs to collect and analyze operating experience are established, the results obtained and the conclusions drawn are acted on and that existing mechanisms are used to share important experience with international bodies and with other operating organizations and regulatory bodies**

(viii) **the generation of radioactive waste resulting from the operation of a nuclear installation is kept to the minimum practicable for the process concerned, both in activity and in volume, and any necessary treatment and storage of spent fuel and waste directly related to the operation and on the same site as that of the nuclear installation take into consideration conditioning and disposal**

The NRC relies on regulations in 10 CFR and internally developed associated programs in granting the initial authorization to operate a nuclear installation and in monitoring its safe operation throughout its life. This section describes the most significant regulations and programs corresponding to each obligation of Article 19. It also includes a discussion on lessons learned from Fukushima.

Question No. 170
Are the requirements for Technical Specifications based on Risk, Deterministic or a combination thereof?
Answer: A brief answer is that Technical Specifications are based on combination of deterministic and risk information.

Before the mid-1980s, Technical Specification requirements were based on deterministic assessments. Since the mid-1980s, the NRC has been reviewing and granting improvements to technical specifications, some of which are based, in part, on PRA. The July 1995 revision to 10 CFR 50.36, “Technical Specifications,” reiterated that the Commission expects licensees to use a plant-specific PRA or risk survey in preparing proposed technical specifications.

In August 1995, the NRC adopted a final policy statement on the use of PRA methods in nuclear regulatory activities that encourages greater use of PRA to improve safety decisionmaking and regulatory efficiency. Since that time, the industry and the NRC have been pursuing increased use of PRA in developing improvements to technical specifications.

In ways consistent with the Commission’s policy statement on technical specifications and the use of PRA, the NRC and the industry continue to develop more fundamental risk-informed improvements to the current system of technical specifications. We use the term “risk management technical specifications” to emphasize the goal of constructing technical specifications that reinforce the proactive management of the total risk presented by the plant configuration and actions that might be needed to respond to emergent conditions. These improvements are intended to maintain or improve safety while reducing unnecessary burden and to bring technical specification requirements into congruence with the Commission’s other risk-informed regulatory requirements. Technical Specifications have taken advantage of risk technology as experience and capability have increased.

Technical Specifications are required by 10 CFR 50.36 and must include:

- safety limits and limiting safety-system settings,
- limiting conditions for operation,
- surveillance requirements,
- design features, and
- administrative controls.

The Technical Specifications are derived from the analyses and evaluation included in the safety analysis report. The bulk of the Technical Specifications are limiting conditions for operation for SSCs that are required by criteria stipulated in 50.36(c)(2)(ii), which states that “A technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the following criteria:

(A) *Criterion 1*. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
(B) *Criterion 2*. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
(C) *Criterion 3*. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
(D) *Criterion 4*. A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.”

The first three criteria are deterministic in nature and the fourth involves risk information. Almost all SSCs in the Technical Specifications are included because of the first three criteria. In general, most of the risk-informed improvements made to date have addressed the required actions that a licensee must take when a limiting condition for operation is not met, adjusted the time licensees have to take corrective action, or allowed licensees to optimize their surveillance frequencies using a risk-informed methodology.

The justifications (deterministic and risk-informed) for the contents of the Technical Specifications are contained in the Bases. The Bases are required by 50.36(a)(1), which states that “A summary statement of the bases or reasons for such specifications, other than those covering administrative controls, shall also be included in the application, but shall not become part of the technical specifications.”

Eight initiatives for fundamental improvements to the standard Technical Specifications have been or are being developed by the industry (through the Risk-Informed Technical Specifications Task Force) and have been discussed with the NRC staff in public meetings:

1. Initiative 1, “Technical Specification Actions End States Modifications”: This initiative permits, for some systems, entry into hot shutdown rather than cold shutdown to repair equipment (approved May 2012);
2. Initiative 2, “Missed Surveillances, Surveillance Requirement 3.0.3”: This initiative permits the extension of up to one surveillance interval of an inadvertently missed surveillance after assessing and managing the risk (approved September 2001);
3. Initiative 3, “Modification of Mode Restraint Requirements of Limiting Condition for Operation 3.0.4 and Surveillance Requirement 3.0.4”: This initiative permits, for most systems, transitioning up in mode with inoperable equipment, relying on compliance with the technical specification actions of the higher mode, after assessing and managing the risk (approved April 2003);
4. Initiative 4b, “Flexible Completion Times”: This initiative permits, contingent on the results of a plant-configuration risk assessment, temporary extension of the existing completion time within a limiting condition for operation using a quantitative implementation of 10 CFR 50.65(a)(4) (approved March 2012);
5. Initiative 5b, “Relocation of Most Surveillance Requirement Frequency Requirements from Technical Specifications to a Licensee-Controlled Program”: This initiative permits most surveillance requirement frequencies to be determined by the licensee through a process defined in a Technical Specifications program (approved July 2009);
6. Initiative 6, “Modification of Limiting Condition for Operation 3.0.3 Actions and Completion Times”: This initiative would provide a completion time for corrective action of up to 24 hours for certain Limiting Conditions for Operation before requiring the default or explicit entry into the Limiting Condition for Operation 3.0.3 shutdown track;
7. Initiative 7, “Non-Technical Specifications Support System Impact on Technical Specifications Operability Determinations”: This initiative permits a risk-informed delay time before entering Limiting Condition for Operation actions for inoperability because of the loss of a support function provided by equipment outside technical specifications; initiative 7 has been applied to snubbers and barriers (approved October 2006);
8. Initiatives 8a and 8b, “Remove/Relocate Non-Safety and Non-Risk Significant Systems from Technical Specifications that Do Not Meet the Four Criteria of 10 CFR 50.36”: Initiative 8a would review technical specifications to remove systems that were included solely because they were judged risk-significant at one time and have now been shown by analysis not to be. Initiative 8b would make the scope of technical specifications depend only on risk significance.

Question No. 171
Does the present development of risk-informed improvements to the technical specifications include insights from lessons learned from Fukushima?
Answer: Possible future Technical Specification enhancements resulting from Fukushima are primarily in the following areas: • Requiring one train of onsite emergency electrical power operable for spent fuel pool makeup and spent fuel pool instrumentation (for water level, temperature, radiation levels) when there is irradiated fuel in the spent fuel pool, regardless of the operational mode of the reactor; • Requiring seismically qualified spent fuel pool makeup water and means to spray water into the spent fuel pools, including requirements on spent fuel pool water parameters; and, • Requiring that for each operating reactor design, Technical Specifications administrative controls cite the approved Emergency Operating Procedures technical guidelines for that plant design. Risk information will be factored into determining whether Technical Specifications are required and the eventual content of those Technical Specifications. As to whether changes are needed to Technical Specification risk-informed improvements, such as the eight "Risk-Informed Initiatives," nothing has yet been determined. As PRAs are improved and cover more modes of operation, such as shutdown operations, this would be reflected in the implementation of the risk-informed initiatives. For example, when PRAs include shutdown modes, one would expect the application of the Risk-informed Completion Time initiative (Initiative 4b) to be expanded.
Question No. 172
Could the USA specify if the 12 recommendations of the Near Term Task Force have already been implemented? If not, what is the deadline?
Answer: In following Commission direction, the NTTF conducted a systematic and methodical review of U.S. NRC processes and regulations to determine whether the agency should make additional improvements to its regulatory system and made recommendations to the Commission for its policy direction. The Commission evaluated the recommendations of the NTTF and subsequently agreed to plans from the NRC staff on how to address the identified issues, as well as other issues that had not been included in NTTF recommendations. In October 2011, the Commission chartered the JLD under the direction of a senior staff steering committee to implement the NRC's longer-term review, including those items identified in the Chairman's tasking memorandum for longer-term review, recommendations for evaluation that were provided by the NTTF and approved by the Commission, and any other review topics as directed by the Commission. The scope of the steering committee's review includes power and non-power reactors, non-operating reactors, and non-reactor NRC licensees and is informed by interactions with external stakeholders. As part of this process, the U.S. is addressing the lessons learned in light of the accident at Fukushima. The status of specific recommendations differ, with some having been implemented, some actively underway, and some still being evaluated. While the Tier 1 issues are expected to be resolved by the end of 2016, several of the NTTF recommendations involve long-term studies for which firm deadlines have not been established.

For the status of the implementation of the lessons learned from the accident at Fukushima, please visit http://www.nrc.gov/reactors/operating/ops-experience/japan-dashboard.html.
Question No. 173
Could the USA specify how many events per year are reported by the licensees to the US NRC, and how many events are reported by the NRC to the IAEA IRS?
Answer: Over the past three years, power-reactor licensees have reported about 650 events each year to the NRC under the requirements of 10 CFR 50.72 (in Event Notifications). Of these, about 400 events a year also receive a more detailed followup report from reactor licensees under the requirements of 10 CFR 50.73 (in Licensee Event Reports). The NRC typically submits between 15 and 20 reports a year to the IAEA Incident Reporting System (IRS). In general, the NRC submits reactor-related generic communications to the Incident Reporting System. In addition, since 2010 the NRC has also submitted Licensee Event Reports describing events of significant regulatory interest.
Question No. 174
A paragraph on inspections and inspection findings is missing (ISI, RPV internals, fire protection). Is this subject reported in a different way?
Answer: The NRC inspection program is discussed in section 6.3.2 of the report. Specific focus areas included in the inspection program are discussed in other sections; fire protection is covered in section 6.3.8 and inservice inspection (ISI) requirements are discussed in sections 10.3.2 and 14.2. Specific operating-experience feedback developed from review of different inspections is not covered in this report.
Question No. 175
What are the results of the operating experience feedback program?
Answer: The NRC's operating-experience program processes information for feedback at varying levels depending on the safety significance and potential generic applicability of the issues. The vast majority of the operating experience, which has very low safety significance, is communicated to the relevant technical groups within the NRC and stored in a database for future trending and analysis. Operating experience with higher safety significance receives evaluation to determine what specific actions might be necessary to reduce the frequency or significance of similar events in the future. Actions resulting from this feedback of operating experience may include a generic communication such as an IN, a revision to inspection procedures, formal briefings with NRC management, discussion with international counterparts, or recommendations to modify the regulations.
Question No. 176
Can you give information on the experience gained from the replacement of the Significant Event Evaluation and Information Network program with the Operating Experience and Construction Experience programs used by INPO?
Answer: INPO replaced the Significant Event Evaluation and Information Network with the Operating Experience and Construction Experience Programs in November 2010. The new programs substantially updated the former program description to take into account the many changes in the industry and in technology since the program's inception. None of the original objectives of the Significant Event Evaluation and Information Network Program or Equipment Performance Information Exchange designers have been altered. Most of the substantive changes were to internal processes at INPO and its members and did not impact the NRC's interaction with the program. The only visible change to the NRC was that the names of the reports changed and construction experience was added. While Significant Event Evaluation and Information Network reports were designated "Significant Operating Experience Reports,"

"Significant Event Reports," "Significant Event Notifications," and "Topical Reports," the names have been clarified so that all reports are now referred to as "INPO Event Reports," with the significance determined by the designation of the report as Level 1 (most significant) through Level 4 (least significant). Internal changes to the program within INPO mean that the NRC has seen an increase in the number of lower-level INPO Event Reports that are published each year compared to the number of Significant Event Evaluation and Information Network reports that were published in any given year. However, the substance of the reports, particularly the reports describing the most significant events, remains essentially unchanged.

Question No. 177

PART3. Industry Response to the accident of Fukushima (INPO)
Regarding support personnel to assist with setup and deployment of post Fukushima equipment, could you please give more details of number of people and qualification needed, as well as procedures to manage dispatching on time qualified technicians from other utilities?

Answer: Each U.S. site is required to conduct a staffing study to determine the staffing necessary to respond to an event simultaneously affecting all units at the site. The study will include support personnel to deploy and operate portable equipment. Detailed staffing numbers will not be available until these studies are complete. Results will set the minimum staffing requirements for initial response to an event and the number and timing of additional personnel to be called in for support.

If additional portable equipment is required from offsite, the equipment will be dispatched from one of two Regional Response Centers or from a nearby station. In general, the Regional Response Center's equipment and the onsite portable equipment are identical. A small number of personnel will accompany the Regional Response Center's equipment to the site and will operate it as directed by site personnel. Personnel from another site could also be called on to travel to the plant experiencing the event. While still under development, identical training is expected to be developed and delivered to Regional Response Centers and to site personnel who will operate the portable equipment. The tasks to be performed are relatively simple and instructions/procedures are being human-factored to avoid confusion and to limit the need for extensive initial and continuing training.

Question No. 178

Can you give insights on the decision to request "reliable hardened containment vents" for BWR Mark I and II as a Tier 1 Recommendation and address other designs of operating NPPs as a Tier 3 Recommendation?

Answer: All U.S. containment designs are encompassed in the lessons learned in light of the nuclear accident at Fukushima (http://pbadupws.nrc.gov/docs/ML1118/ML111861807.pdf).
A reliable hardened vent for Mark I and Mark II containments (Recommendation 5.1) is a Tier 1 recommendation for which actions are to begin without unnecessary delay. A reliable hardened vent for other containment designs (Recommendation 5.2) is a Tier 3 recommendation that requires longer-term evaluation. The NRC will use insights from the development and implementation of requirements for the Mark I and Mark II containments to inform its evaluations of what, if any, changes should be proposed for other containment designs.

Licensees of nuclear power plants with BWR Mark I and Mark II containments must install hardened vents able to function following a severe accident. In addition, the NRC is evaluating possible regulations that would require each licensee to develop a filtering strategy and improve accident-management capabilities for maintaining containment integrity. Engineered filtering systems are one of the options being evaluated as part of the rulemaking process.

For more information, please visit http://www.nrc.gov/reactors/operating/ops-experience/japan-dashboard/hardened-vents.html.
Question No. 179
Can you explain in more detail the requirements for “ensuring reliable hardened containment vents” for BWR Mark I and II designs? Is there a USNRC requirement on mitigating the release of radioactive substances (by on-site equipment) during beyond design basis accidents?
Answer: In SRM-SECY-12-0157, the Commission approved “Order Modifying Licenses with Regard to Reliable Hardened Containment Vents,” which required licensees of BWRs with Mark I and Mark II containments to upgrade or replace the reliable hardened vents required by Order EA-12-050 with a containment venting system designed and installed to remain functional during severe accident conditions. The Commission also directed the NRC staff to develop technical bases and a possible rulemaking for filtering strategies with drywell filtration and severe-accident management of BWR Mark I and II containments; for this rulemaking, staff is to consider design and installation of an engineered filtered containment venting system intended to prevent the release of significant amounts of radioactive material following the dominant severe-accident sequences at BWRs with Mark I and Mark II containments. The Commission directed the staff, in addition, to develop requirements and technical acceptance criteria for confinement strategies and requirements for licensees to justify operator actions and systems or combination of systems (such as suppression pools, containment sprays, and separate filters) to accomplish the confinement function and meet the requirements. The NRC rulemaking process will evaluate the potential costs and benefits of equipment and strategies to reduce the release of radioactive materials beyond the protections provided by NRC requirements intended to prevent core melt and contain radioactive materials during a severe accident. SRM-SECY-12-0157 can be viewed at http://www.nrc.gov/reading-rm/doc-collections/commission/srm/2012/2012-0157srm.pdf SECY-12-0157 can be viewed at http://www.nrc.gov/reading-rm/doc-collections/commission/secys/2012/2012-0157scy.pdf.
Question No. 180
The Near-Term Task Force concluded that enhancements to safety and emergency preparedness were warranted. When have these enhancements to be implemented? Are exceptions possible or will delays in implementing of these enhancements have influence on licensing and operation?
Answer: For up-to-date information on emergency-preparedness staffing and communications enhancements, please visit http://www.nrc.gov/reactors/operating/ops-experience/japan-dashboard/emerg-preparedness.html. Concerning the Tier 1 activity on emergency-preparedness staffing and communication, the NRC has approved industry guidelines for assessing beyond-design-basis accident response staffing and communications capabilities (see http://pbadupws.nrc.gov/docs/ML1212/ML12125A412.pdf). Concerning Tier 3 activities related to emergency preparedness: In SECY-12-0095 (ADAMS Accession No. ML12165A092), the following four Tier 3 items were included within one program plan: • Emergency-preparedness enhancements for prolonged SBO and multiunit events; • Emergency Response Data System capability;

- Additional emergency-preparedness topics for prolonged SBO and multiunit events; and
- Emergency-preparedness topics for decisionmaking, radiation monitoring, and public education.

These four items collectively originated from NTTF Recommendations 9.1 through 9.3, 10.1 through 10.3, and 11.1 through 11.4. The program plan outlined in SECY-12-0095 described an approach to collectively address these items using an Advance Notice of Proposed Rulemaking, which is a tool that allows the NRC to solicit early written stakeholder input on a new potential rulemaking effort. The staff still intends to take this approach and expects to use the Advance Notice of Proposed Rulemaking feedback to help determine whether there is a need for rulemaking and, if so, what the scope and content of the rulemaking should be. The staff now expects to issue the Advance Notice of Proposed Rulemaking in FY 2016.

Question No. 181

When will the final version of the spent fuel pool study be expected? Does a deadline exist?

Answer: The spent fuel pool study, provided as an enclosure to SECY-13-0112, ADAMS Accession No. ML13256A342, was issued on October 9, 2013 (http://www.nrc.gov/reading-rm/doc-collections/commission/secys/2013/2013-0112scy.pdf).

Question No. 182

Is there an USNRC requirement/regulation for assistance to an accidently affected NPP site/area from outside the affected area? Would such assistance from other parts of the country cover hardware and personal? Is air-borne assistance arranged?

Answer: Order EA-12-049 requires a three-phase approach to mitigating beyond-design-basis events, with an initial response phase using installed equipment, a transition phase using portable equipment and consumables to provide core and spent fuel pool cooling and maintain the containment functions, and a third phase of indefinite sustainment of these functions using offsite resources. Maintenance of core and spent fuel pool cooling and containment functions requires the overlap of the initiating times for the phases with the licensee's performance times for the prior phases. The nuclear industry is creating two Regional Support Centers and each licensee is required to develop plans for the delivery of equipment and supplies to the affected reactor site through a variety of methods.

Question No. 237

Please explain what kind of information is collected and stored in operating experience database as lower level operating experience.

Answer: The operating experience group reviews low-level issues and events reported in Event Notifications and Licensee Event Reports reported to the NRC in accordance with 10 CFR 50.72 and 50.73 respectively. In addition, the database contains information on selected events of interest at U.S. plants that plant inspectors review during daily phone calls with the Regions and the operating experience group. These events are tracked for future trending and analysis.

NRC FORM 335 (12-2010) NRCMD 3.7 U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET** *(See instructions on the reverse)*	1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.) NUREG-1650 Revision 5, Supplement 1
2. TITLE AND SUBTITLE Answers to Questions form the Peer Review by Contracting Parties on the United States of America Sixth National Report for the Convention on Nuclear Safety	3. DATE REPORT PUBLISHED MONTH: November YEAR: 2014 4. FIN OR GRANT NUMBER
5. AUTHOR(S) U.S. Nuclear Regulatory Commission (NRC) Institute of Nuclear Power Operations (INPO)	6. TYPE OF REPORT Technical 7. PERIOD COVERED (Inclusive Dates) October 2013-March 2014

8. PERFORMING ORGANIZATION - NAME AND ADDRESS (If NRC, provide Division, Office or Region, U. S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

Division of Inspection and Regional Support
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS (If NRC, type "Same as above", if contractor, provide NRC Division, Office or Region, U. S. Nuclear Regulatory Commission, and mailing address.)

Same as above (item 8)

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

The Convention on Nuclear Safety (CNS) was adopted in June 1994 and entered into force in October 1996. The objectives of the CNS are to achieve and maintain a high level of nuclear safety worldwide. Contracting parties to the Convention have four obligations: submit a national report for peer review, review the national reports of other contracting parties, respond to questions and comments submitted by the contracting parties, and participate in the organizational and review meetings. The United States published its sixth national report for peer review in October 2013 (NUREG 1650, "The United States of America National Report for the Convention on Nuclear Safety: Sixth National Report, October 2013," Revision 5). Addendum 5 to NUREG 1650 documents the answers to questions raised by contracting parties during their peer reviews of the U.S. national report. Specifically, the questions and answers resulting from the peer reviews concern the safety of existing nuclear installations, the legislative and regulatory framework, the regulatory body, responsibility of the licensee holder, priority to safety, financial and human resources, human factors, quality assurance, assessment and verification of safety, radiation protection, emergency preparedness, siting, design and construction, and operation, and implementation of the lessons learned from the Fukushima accident. The sixth review meeting of the CNS was held at the International Atomic Energy Agency in Vienna, Austria, from March 24 through April 4, 2014.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)	13. AVAILABILITY STATEMENT
Convention on Nuclear Safety, CNS, nuclear plants, programs, challenges, reviews, legislation, regulation, reactor oversight, ROP, public participation, probabilistic risk assessment, PRA, performance-based, quality, safety, knowledge management, human capital, security, siting, design, construction, operations, new reactors, radiation protection, emergency preparedness, financial, human factors, INPO, Institute of Nuclear Power Operations, peer review, international, IAEA, International Atomic Energy Agency, Contracting Parties	unlimited 14. SECURITY CLASSIFICATION *(This Page)* unclassified *(This Report)* unclassified 15. NUMBER OF PAGES 16. PRICE

Printed on recycled paper

Federal Recycling Program

NUREG-1650, Rev. 5
Supplement 1

Answers to Questions from the Peer Review by Contracting Parties on the United States of America Sixth National Report for the Convention on Nuclear Safety

November 2014