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NRC Report
February 2013 - Los Angeles, CA

1. Amendment to 10 CFR 50.55a – ASME Code Edition/Addenda

ASME Edition/Addenda

The NRC has approved Section III, Division 1, and Section XI, Division 1, of the *Boiler and Pressure Vessel Code* through the 2008 Addenda (76 FR 36232).

Proposed Rule – 2009 through 2011 Edition/Addenda

The proposed rule that would incorporate the 2009 Addenda, the 2010 Edition, and the 2011 the Addenda is scheduled for publication in summer 2013 (hereafter called the edition rulemaking). The NRC staff is currently revising the proposed edition rulemaking to be consistent with the new Office of the Federal Register (OFR) guidelines addressing the format of rulemakings and also to be consistent with the changes that will be proposed in the rulemaking addressing the incorporation into 10 CFR 50.55a by reference of the revised ASME Code Case regulatory guides (hereafter called the Code Case rulemaking (see No. 2 below)). The proposed Code Case rulemaking is the lead rulemaking effort with respect to addressing the new OFR guidelines. Accordingly, the proposed edition rulemaking will be published after the proposed Code Case rulemaking.

2. ASME Code Case Rulemaking/Regulatory Guides

The latest NRC-approved versions of the ASME Code Case Regulatory Guides (RG) are: RG 1.84, Revision 35, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III," and RG 1.147, Revision 16, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," (75 FR 61530).

Draft Revision 36 to RG 1.84, draft Revision 17 to RG 1.147, draft Revision 1 to RG 1.192, "Operation and Maintenance Code Case Acceptability, ASME OM Code," and draft Revision 4 to RG 1.193, "ASME Code Cases Not Approved for Use," are scheduled to be published for public comment in February 2013 in conjunction with the proposed Code Case rulemaking. The draft guides will address Supplements 1 through 10 to the 2007 Edition. *The risk-informed Code Cases in draft RG 1.192 have been clarified to make them consistent with current generic NRC risk-informed guidance.* The issues raised by Raymond A. West in a petition for rulemaking dated December 14, 2007, and revised on December 19, 2007, will be addressed in the proposed Code Case rulemaking. The proposed Code Case rulemaking and guides are scheduled for publication in spring 2013.

Comment [GLS1]: I'm not sure I labeled these rulemakings correctly -- they are confusing !!

The NRC staff has completed its review of Supplements 1 – 11 to the 2007 Edition and Supplement 0 through Supplement 8 to the 2010 Edition. Supplement 9 to the 2010 Edition is currently under review.

Draft Revision 37 of RG 1.84, Ddraft Revision 18 of RG 1.147, and Ddraft Revision 5 of RG 1.193 have been initiated. These drafts will address the Code Cases published by the ASME in Supplement 11 to the 2007 Edition through Supplement 8 to the 2010 Edition. It is expected that these drafts will be published in the *Federal Register* for comment within a few months after

Revision 36 of RG 1.84, Revision 17 of RG 1.147, and Revision 4 of RG 1.192 are published as final guides.

3. ASME Certification Mark

The ASME representatives at the August 17, 2012, NRC/ASME management coordination meeting requested that the NRC address the regulatory transition from the Code Symbol Stamps (CSS) to a Certification Mark (CM) in a timely manner as licensees would be ordering replacement equipment with the CM in the near-term. Licensees have indicated to the ASME their concerns with respect to whether or not there are potential licensing implications.

NRC technical staff met with NRC's legal staff from the Office of the General Counsel (OGC) to discuss this issue. OGC understands that the ASME has determined the technical equivalency of CSSs and the CM. However, OGC indicates that this is a legal issue, i.e., edition and addenda earlier than the 2011 Addenda specifically require CSSs, and the interpretation by the ASME of equivalency cannot address an issue which is specific to 10 CFR 50.55a.

The NRC developed a regulatory path forward to address the issue. The NRC will issue a Regulatory Issue Summary (RIS) in conjunction with an Enforcement Guidance Memoranda (EGM) (i.e., exercise enforcement discretion) in late February/early March 2013. In the meantime, for the two new-reactor plants under construction and any operating facilities that are in a more urgent situation, the NRC will authorize, on a plant-specific basis, an alternative pursuant to 10 CFR 50.55a(a)(3) to allow the use of the CM until the final rule incorporating by reference Section III, 2011 Addenda to the ASME Boiler and Pressure Vessel Code, is issued by the NRC. The NRC plan for addressing the CM issue is described in a letter from R.W. Borchardt (NRC) to B. Erler and R. Swayne (ASME) dated October 2, 2012 (ADAMS Accession No. ML12262A005).

4. Risk-Informed Activities

The NRC issued its final Safety Evaluation (SE) on the Electric Power Research Institute (EPRI) Topical Report (TR) 1021467 (previously numbered 1018427), "Nondestructive Evaluation: Probabilistic Risk Assessment Technical Adequacy Guidance for Risk-informed Inservice Inspection Programs," on January 18, 2012 (available on the NRC's public Web site in the Agencywide Documents Access and Management System (ADAMS), Accession Nos. ML11325A375 (cover letter) and ML11325A340 (final SE)). The TR was developed, in part, as a technical support document for Code Case N-716, "Alternative Piping Classification and Examination Requirements, Section XI, Division 1." The SE contained four conditions and four plant-specific action items associated with use of the TR, discussed in Section 4.0 of the SE. The NRC requested that EPRI publish a revised version of the TR incorporating the changes outlined in the SE. A letter from EPRI to NRC dated June 18, 2012 (ML12171A450) provided the updated version of EPRI Report 1021467.

It is expected that Code Case N-716-1 will be approved without condition in a future revision of Regulatory Guide 1.147. Comments provided to Section XI by NRC staff during consideration of the revisions to Code Case N-716 addressed concerns with the application of risk-informed

inservice inspection in new construction. It should be noted that new plants will be required to submit an application to use the Code Case.

In a separate matter, by letter dated January 17, 2013 (ML12362A354), the NRC notified the Pressurized Water Reactor Owners Group (PWROG) that the NRC staff had completed its review of Topical Report (TR) WCAP-17236-NP-A, Revision 0, "Risk-Informed Extension of the Reactor Vessel Nozzle Inservice Inspection Interval," and that the staff had determined that the subject version is acceptable for referencing in licensing applications for nuclear power plants to the extent specified and under the limitations delineated in this TR.

5. Generic Activities on Material Degradation/PWR Alloy 600/182/82 PWSCC

North Anna 1

On March 24, 2012, during the Alloy 600 dissimilar metal weld overlay work on the 'B' Reactor Coolant loop hot leg to the 'B' Steam Generator nozzle weld at North Anna 1, significant defects were identified by leakage that occurred during weld excavation. Approximately 1 inch of weld material was removed prior to the seepage being identified.

The NRC considers this to be a serious incident in that an examination was recently conducted of this weld, and ~~none~~ one of the five indications were detected during the examination. It was determined that two of the cracks were greater than 80 per cent through-wall and three were greater than 40 per cent through-wall. Teams were dispatched to the site on two separate occasions, including nondestructive examination (NDE) experts, to gather additional information.

The NRC held a public meeting on May 30, 2012, at which industry representatives discussed the formulation of a team to investigate the impact of the occurrence and determine appropriate industry corrective actions. A subsequent meeting was held on July 20, 2012, at NRC headquarters where industry representatives ~~discussed~~ identified 29 corrective actions that were initiated. A follow-up public meeting with industry representatives was held on September 11, 2012, where progress-to-date on the 29 actions was discussed.

The NRC conducted an assessment of the event and reviewed the licensee's root cause analysis. A Pacific Northwest National Laboratory Technical Letter Report (PNNL-21546) developed in concert with the NRC staff entitled, "Evaluation of Manual Ultrasonic Examinations Applied to Detect Flaws in Primary System Dissimilar Metal Welds at North Anna Power Station," is available (in ADAMS (Accession No. ML12200A216)). The NRC and EPRI are currently conducting independent evaluations with respect to the probe design and characteristics, and mockup design. The NRC is also assessing factors such as method of examination, use of site specific mockups, and adequacy of PDI technical justification. Subsequent reports are planned that will discuss the results of on-going research, and the status of the open issues identified in PNNL-21546.

The next NRC/NDE Implementation Focus Group (NIFG) meeting ~~will be~~ is tentatively held scheduled ~~for~~ on Thursday, February 28, 2013, at NRC headquarters in Rockville, MD. A public meeting notice ~~with agendas~~ for these meetings will be issued once the details are finalized.

Dissimilar Metal Welds

In 2006, ASME started the development of a Code Case for inspection of Alloy 82/182 butt welds. On January 26, 2009, Code Case N-770, "Alternative Examination Requirements and Acceptance Standards for Class 1 PWR Piping and Vessel Nozzle Butt Welds Fabricated with UNS N06082 or UNS W86182 Filler Material With or Without Application of Listed Mitigation Activities, Section XI, Division 1," was approved by the ASME to address inspection of these welds, and the NRC included the Code Case in the recently published final amendment to 10 CFR 50.55a (see #1 above). Revision 1 of Code Case N-770 was approved by ASME on 12/25/2009. NRC held a public meeting on July 12, 2011 to discuss implementation of ASME Code Case N-770-1 under 10 CFR 50.55a(g)(6)(ii)(F). The slides for the meeting are in ADAMS under Accession No. ML11189A220. An NRC meeting summary, published on August 12, 2011, is available in ADAMS under Accession No. ML112240818 that includes 48 Questions & Answers that address the questions raised during and after the meeting that may assist licensees when implementing 50.55a(g)(6)(ii)(F).

The NRC staff continues to monitor and evaluate operating experience to ensure that the current inspection schedules are adequate.

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Other items:

The NRC has reviewed the information on Gravelines, and the NRC will be requesting or has requested ????? that the ASME to initiate Code action to develop requirements for volumetric and/or surface examination of the bottom head penetrations and associated welds. This action should include the development of inspection frequency, qualification requirements and consideration for mitigation/repair activities.

Five cold head plants have now experienced cracking of top head penetration nozzles and/or associated welds. The NRC is evaluating the need for changes to the inspection periodicity for cold head plants. Specifically, the need to increase bare metal visual examinations to ensure timely detection of leakage through J-groove welds. Cold head plants should highly consider submitting proactive repair requests prior to their refueling outages.

Five cracks were missed during a dissimilar metal (DM) weld inspection at a plant. Questions were raised regarding the adequacy of current inspection and qualification requirements for performing inspections on DM welds for the detection of primary water stress corrosion cracking (PWSCC). If used, the site-specific mockup process must be robust and meet the letter of Supplement 10 of Appendix VIII. Currently, there is no mechanism in the ASME code or 10CFR50.55a to allow the use of site-specific mockups to modify essential variables. In accordance with Paragraph 4(d) of Supplement 10, "to qualify new values of essential variables, at least one personnel performance demonstration set is required."

When issues occur with un-encoded exams, the lack of recorded/reviewable examination results makes diagnosis of problems difficult or impossible. The NRC is considering new requirements on ASME Code Case N-770-1 to require encoded examinations of PWSCC susceptible dissimilar metal butt welds and mitigated welds with indications of PWSCC in the susceptible material.

6. New Reactor Licensing Activities

As of January 31, 2013, the status of new reactor licensing under 10 CFR Part 52 is as follows:

Design Certification

NRC has issued four design certifications to-date (ABWR, System 80+, AP600, and AP1000). These are certified in 10 CFR Part 52, Appendices A, B, C, and D, respectively.

The NRC is currently reviewing two design certifications:

- AREVA's EPR (evolutionary pressurized-water reactor design from France)
- Mitsubishi Heavy Industries' US-APWR (advanced pressurized water reactor design from Japan)

The NRC staff completed its review of General Electric-Hitachi's ESBWR (first passive BWR) and issued its final safety evaluation report (FSER) in March 2011. On March 24, 2011, the NRC issued in the *Federal Register* a proposed rule (76 FR 16549) for public comment on the ESBWR design certification. Five public comments were received on the proposed rule. The final rule to certify General Electric-Hitachi's ESBWR standard plant design in Appendix E to 10 CFR Part 52 is on hold pending resolution of steam dryer issues.

The NRC staff completed its review of Westinghouse's AP-1000 design certification amendment and issued its FSER in August 2011. On February 24, 2011, the NRC issued in the *Federal Register* a proposed rule (76 FR 10260) for comment on the AP-1000 design certification amendment. Approximately 13,500 comments were received on the proposed rule. The NRC issued a final rule amending Westinghouse's AP-1000 design certification rule (Appendix D to 10 CFR Part 52) on December 30, 2011.

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In addition, the NRC staff is reviewing two applications for design certification renewal:

- ABWR GE-Hitachi (application submitted on December 7, 2010)
- ABWR GE-Toshiba (Revision 1 to application submitted on June 22, 2012)

Early Site Permits (ESPs)

NRC has issued four ESPs (Clinton, Grand Gulf, North Anna, and Vogtle). The NRC's issuance of the Vogtle ESP on August 26, 2009, was the first based on a specific technology (AP-1000) and the first to include a limited-work authorization (LWA).

The NRC received an application for an ESP for the Victoria County Station submitted by Exelon on March 25, 2010. The site is located in Victoria County, Texas, with no specific technology selected. On August 28, 2012, Exelon requested withdrawal of the Victoria County Station ESP application ~~be withdrawn~~ from the docket. By letter dated October 3, 2012, NRC accepted the applicant's request, and the application was withdrawn.

The NRC received an ESP application for the PSEG site in New Jersey (same site as Hope Creek and Salem 1&2). The ESP application was tendered on May 25, 2010, and was docketed on August 4, 2010. This application uses the Plant Parameter Envelope (PPE) approach which means no specific reactor design has been selected.

Combined License (COL) Applications

NRC is currently reviewing 10 COL applications (24 new reactor units):

- 1 ABWR South Texas Project 3 and 4
- 4 AP-1000 William S. Lee Station 1&2, Shearon Harris 2&3, Levy County 1&2, Bellefonte 3&4*, and Turkey Point 6&7
- 1 ESBWR Fermi 3, Grand Gulf 3*, River Bend 3*, Victoria County 1 and 2**
- 2 EPR Calvert Cliffs 3, Nine Mile Point 3*, Bell Bend, Callaway 2*
- 2 US-APWR Comanche Peak Units 3 and 4, North Anna 3

* NRC staff review suspended at request of applicant.

** Application withdrawn

The NRC staff completed its review of the Vogtle Electric Generating Plant, Units 3&4 COL application and issued its final safety evaluation report (FSER) on August 5, 2011. The Vogtle 3&4 plants reference the AP-1000 design certification amendment. A mandatory hearing conducted by the Commission was held on September 27-28, 2011. Following Commission affirmation, the NRC issued the combined license and limited work authorization for Vogtle 3&4 on February 10, 2012. It was the first combined license issued by the NRC to construct and operate a nuclear power plant under the alternative licensing process in 10 CFR Part 52. It is the first time since 1978 that the NRC issued a license to construct a nuclear power plant in the United States.

The NRC staff completed its review of the V.C. Summer 2&3 COL application and issued its FSER on August 17, 2011. The V.C. Summer 2&3 plants reference the AP-1000 design certification amendment. A mandatory hearing was conducted by the Commission on October 12-13, 2011. Following Commission affirmation, the NRC staff issued the combined license for V.C. Summer 2&3 on March 30, 2012.

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Advanced Reactors Program

NRC established an advanced reactors program in the Office of New Reactors. Currently, there are no applications under review, but several applications are expected in the next three years including:

- **High Temperature Gas-Cooled Reactors:**
 - Next Generation Nuclear Plant (DOE) – Pre-application interactions are continuing. The Secretary of Energy sent letters to Congress on October 17, 2011, describing his decision regarding continuation of the NNGP project. The Secretary informed Congress that the project will continue high-temperature-reactor research and development activities, interactions with the NRC to develop a licensing framework, and efforts to form a cost-shared public-private partnership. However, initial NNGP design parameters have not been selected, pending establishment of the public-private partnership.
- **Integral PWRs (iPWRs):**
 - NuScale (iPWR) – NuScale Power is developing a modular, scalable 45 MWe iPWR. Pre-application reviews are currently under discussion. In addition, NuScale LLC is working together with NuHub to pursue a small modular project (SMR) project at the Savannah River Site in South Carolina. The team's plan is to assist in the design certification licensing process and development of the reference combined-license application (R-COLA).
 - B&W mPower (iPWR)– B&W is developing a modular, scalable 125 MWe iPWR. Pre-application interactions are underway. A standard design certification application expected by the end of 2013. An integrated test facility is under construction. TVA Clinch River is planning to build an mPower plant under the 10 CFR Part 50 licensing process. B&W will pursue a standard design certification under the alternative 10 CFR Part 52 licensing process in parallel when the design details are more finalized. On November 20, 2012, the U.S. Department of Energy announced that they selected the Generation mPower Team (B&W, Bechtel, and TVA) as the recipient of cost-share funding to design and build an SMR in the United States.
 - Westinghouse is developing a modular iPWR design.

The New Reactor Licensing public web-site <http://nrr10.nrc.gov/NRO/new-rx-status/index.cfm> has a list of expected new nuclear power plant applications, and an estimated schedule by fiscal year for new reactor licensing applications.

NRO Vendor Inspection

The NRO vendor inspection program is described in Inspection Manual Chapter (IMC) 2507, "Construction Inspection Program, Vendor Inspection." This IMC will be implemented by various Inspection Procedures (IPs) including:

- IP 43002: Routine Inspections of Nuclear Vendors;
- IP 43003: Reactive Inspections of Nuclear Vendors;
- IP 43004: Inspection of Commercial-Grade Dedication Programs;

IP 43005: NRC Oversight of Third Party Organizations Implementing Quality Assurance Requirements; and
IP 36100: Inspection of 10 CFR Parts 21 and Programs for Reporting Defects and Noncompliance.

FY 13 Vendor Inspection Plans

- Commercial grade dedication organizations
- Manufacturing for valves (all new reactor Design Centers and for operating reactors)
- AP-1000 modular construction
- AP1000 and operating reactor equipment qualification test programs
- AP1000 manufacturing
- Digital Instrumentation and Control for AP-1000 and US-APWR
- Instrumentation and Control for operating reactors

Vendor Inspection Reports Completed, Issued and Planned Inspections

- Nuclear Logistics, Inc, Fort Worth, TX - issued
- Flowserve, Raleigh, NC - issued
- Target Rock, Farmingdale, NY - issued
- Shaw Modular Solutions, Lake Charles, LA - issued
- Flowserve Limatorque, Lynchburg, VA - issued
- Enertech & Utah State University, Brea, CA and Logan UT - issued
- Westinghouse Electric Company, Cranberry Township, PA - issued
- Mitsubishi Heavy Industries, Kobe, Japan - issued
- Mangiarotti, Monfalcone, Italy - issued
- Cives Steel Company, Thomasville, GA - completed
- ABB Inc., Florence, SC - completed
- DRS Consolidated Controls Inc, Danbury, CT - completed
- Weir Valves & Controls USA Inc., Ipswich, MA - completed
- Westinghouse Electric Company, Cranberry Township, PA - ongoing
- Stern Labs, Hamilton, Canada - scheduled
- Scientech, Idaho Falls, ID - scheduled
- QualTech NP, Cincinnati, OH - scheduled
- Flowserve Pumps, Vernon, CA - scheduled

Vendor Inspections continue to identify findings related to commercial grade dedication activities and inadequate Part 21 programs for evaluating and reporting of defects that could cause a substantial safety hazard.

On January 24, 2013, the NRC held a public meeting to discuss the NRC staff's draft regulatory basis to clarify 10 CFR Part 21. The goal of this regulatory basis was to simplify and clarify the rule language in Part 21, provide consolidated regulatory guidance on compliance with Part 21, and enhance regulatory stability and predictability for the entities to which Part 21 applies. A summary of the meeting will be published.

Previously issued NRC inspection and trip reports can be located at:

<http://www.nrc.gov/reactors/new-reactors/oversight/quality-assurance/vendor-insp.html>

Multinational Design Evaluation Program (MDEP) Activities

A. Vendor Inspection Cooperation Working Group (VICWG)

The MDEP VICWG members continue to allow opportunities for NRC staff participation and observation of vendor inspections conducted by regulatory authorities from other countries and for opportunities where participation and observation of NRC vendor inspections by representatives of regulatory authorities from other countries is possible. VICWG objectives include: explore international regulators' vendor oversight requirements and programs; apply lessons learned; exchange vendor inspection insights; and identify areas where international cooperation can yield tangible benefits.

The VICWG is evaluating the advantages and disadvantages of conducting multinational inspections. It is recognized that only specific vendors who routinely apply both GS-R-3, ISO 9000+, and Appendix B requirements would be good candidates for such inspections. The scope and requirements for such inspections would need to be clearly outlined and agreed upon by the VICWG participants.

On November 13-15, 2012, NRC staff participated at the VICWG meeting in Paris, France. The MDEP VICWG continues to achieve its short-term goals and is making progress towards achieving its long-term program goals. Multinational inspections of vendors according to common Quality Assurance/Quality Management (QA/QM) requirements remains a long term goal of the group although there are obstacles to achieving this goal. For the intermediate term, emphasis will be placed on maximizing information sharing, joint inspections (multiple nations inspecting to the regulatory requirements of one country), and witnessing of other regulators' inspections. The VICWG will also continue to work with the Standards Development Organizations (SDOs) to encourage and explore harmonization of QA/QM standards.

The NRC is currently planning a joint multinational inspection with 2 inspectors from KINS of QualTech NP of Curtis-Wright Flow Control Company for March 18-22, 2013.

The next VICWG meeting is tentatively planned for April 23 - 25, 2013 in Paris, France. Representatives from IAEA, relevant standards development organizations (SDOs) and ISO will be invited to give input on potential QA requirements and standards harmonization.

B. Codes and Standards Working Group (CSWG)

MDEP is a multinational initiative to develop innovative approaches to leverage the resources and knowledge of mature, experienced national regulatory authorities who are tasked with the regulatory design review of new reactor plant designs. One of the issue-specific working groups

established under the MDEP organization is the Codes and Standards Working Group (CSWG) whose goal is to achieve harmonization of code requirements for pressure-boundary components.

Harmonizing pressure-boundary codes used by member countries ensures a consistent level of quality and safety in the design of pressure-boundary components such as the reactor vessel, piping, pumps, and valves, and allows components manufactured in other countries to be used in member countries with a relatively minor review and reconciliation of code differences. Such an approach would significantly simplify the licensing of nuclear power plants and reduce the burden on the regulatory authorities on an international scale.

The MDEP/CSWG worked with standards development organizations (SDOs) from several countries (i.e., U.S., Japan, Korea, France, Canada, and the Russian Federation) for the past 5 years to compare each countries' pressure-boundary code requirements for Class 1 vessels, piping, pumps and valves to the requirements of the ASME Boiler and Pressure Vessel Code, Section III. Similarities and differences were documented in a database table and summarized in a final report. The code-comparison effort is the first step to achieve harmonization of pressure-boundary codes and standards. The final code-comparison report and tables were completed for Class 1 vessels, piping, pumps and valves for Korea, Japan, Canada and France and were supplemented with Russia's input in December 2012. Revision 1 to the SDOs' final report, ASME Report STP-NU-051-1, "MDEP-CSWG Code Comparison," and tables were issued as an ASME Standard Technical publication and is publicly available free for download from the ASME Standards Technology, LLC website at http://stllc.asme.org/News_Announcements.cfm.

The MDEP/CSWG issued letters to: ASME, AFCEN (France), JSME (Japan), CSA (Canada), the KEA (Korea), and NIKIET (Russia) SDOs, in part, to ask these SDOs to address their plans to preclude future divergences of code requirements. In response, the SDOs met with MDEP/CSWG in September 2012 and discussed the establishment of an SDO Convergence Board that would pursue convergence of code requirements where realistic and practical, limit further divergence of code requirements, and develop a process for reconciling different code requirements. The SDO Convergence Board is also working with World Nuclear Association's CORDEL Group to identify potential areas of code convergence. The SDO Convergence Board held its first meeting in conjunction with the ASME Boiler Code Week in Washington DC in August 2012.

Public Meetings

A public meeting was held at NRC headquarters on February 5, 2013, to discuss specific aspects of the Commission's Staff Requirements Memorandum to SECY-12-0081, "Risk-Informed Regulatory Framework for New Reactors." The NRC staff discussed use of risk-informed inservice inspection and the applicability of 10 CFR 50.69 to new reactors licensed under 10 CFR Part 52. A summary of the meeting will be published.

A public conference call will be held on Tuesday, February 19, 2013, from 1:00 p.m. to 2:00 p.m. to facilitate communication between the NRC and industry on risk-related topics for new reactors, specifically in the areas of risk-informed in-service inspection and risk-informed safety

classification (10 CFR 50.69). The meeting notice and agenda are available in ADAMS under ML13018A213.

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7. License Renewal Activities

Following are on-going activities related to license renewal:

Current status of applications, staff reviews and approvals

- 73 units approved
- 10 applications (15 units) under review:
 - 1 (2 units) in hearings (Indian Point 2 & 3)
 - 1 (2 units) completed ACRS Full Committee (Limerick 1 & 2)
 - 3 (4 units) awaiting ACRS Full Committee (Davis-Besse [3/2013], Crystal River 3 [schedule TBD, as impacted by containment concrete issues] and Diablo Canyon 1 & 2 [schedule TBD as applicant has requested that renewed license issuance be delayed pending 3-D seismic studies, which is expected prior to December 2015])
 - 1 (1 unit) completed initial ACRS Subcommittee (Seabrook [7/2012] - schedule impacted by follow-up meeting on alkali-silica reaction (ASR) issues)
 - 3 (4 units) awaiting ACRS Subcommittee (Grand Gulf [3/2013], Callaway [5/2013], South Texas Project 1 & 2 [TBD])
 - 1 (2 units) application received (Sequoyah 1 & 2).
- 1 application (4 units) with scheduled submittal date before end of 2013:
 - June 2013 – Byron 1 & 2 and Braidwood 1 & 2
- Other scheduled applications:
 - January to March 2014 – Waterford 3
 - April to June 2014 – Fermi 2
 - October to December 2014 – River Bend
 - January to March 2015 – LaSalle 1 & 2
 - September 2015 – Perry
 - July to September 2017 - STARS
 - January to March 2017 – Clinton
 - October to December 2018 - STARS

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Seventeen units have entered the operating period beyond 40 years:

- Oyster Creek – April 9, 2009
- Nine Mile Point Unit 1 – August 22, 2009
- Ginna – September 18, 2009

- o Dresden Unit 2 – December 22, 2009
- o H.B. Robinson – July 31, 2010
- o Monticello – September 8, 2010
- o Point Beach Unit 1 – October 5, 2010
- o Dresden Unit 3 – January 12, 2011
- o Palisades – March 24, 2011
- o Vermont Yankee – March 21, 2012
- o Surry Unit 1 – May 25, 2012
- o Pilgrim – June 8, 2012
- o Turkey Point Unit 3 – July 19, 2012
- o Quad Cities Unit 1 – December 14, 2012
- o Quad Cities Unit 2 – December 14, 2012
- o Surry Unit 2 – January 29, 2013
- o Oconee Unit 1 – February 6, 2013

Nine additional units will enter the operating period beyond 40 years before the end of 2013:

- o Point Beach Unit 2 – March 8, 2013
- o Turkey Point Unit 4 – April 10, 2013
- o Peach Bottom Unit 2 – August 8, 2013
- o Fort Calhoun Unit 1 – August 9, 2013
- o Prairie Island Unit 1 – August 9, 2013
- o Indian Point Unit 2 – September 28, 2012
- o Oconee Unit 2 – October 6, 2013
- o Brown Ferry Unit 1 – December 20, 2013
- o Kewaunee – December 21, 2013

Comment [GL55]: Are you sure???? Or, shutdown???

Technical Issues

- Final LR-ISG-2011-01: Aging Management of Stainless Steel Structures and Components in Treated Borated Water
 - Issued May 11, 2012.
 - ADAMS Accession No. ML12034A047
- Final LR-ISG-2011-02: Aging Management Program for Steam Generators
 - Issued December 1, 2011.
 - ADAMS Accession No. ML11297A085
- Final LR-ISG-2011-03: Generic Aging Lessons Learned (GALL) Report Revision 2 AMP XI.M41, "Buried and Underground Piping and Tanks"
 - Issued August 2, 2012.
 - ADAMS Accession No. ML12138A296
- Draft LR-ISG-2011-04: Updated Aging Management Criteria for Reactor Vessel Internal Components of Pressurized Water Reactors
 - Issued for comment on March 20, 2012; comment period ended May 21, 2012.
 - ADAMS Accession No. ML12004A149

- Revision 1 of the safety evaluation on the MRP-227 report was issued on December 16, 2011 (ML11308A770).
 - MRP-227-A issued on January 9, 2012.
 - Implements changes to GALL Revision 2 aging management review line items and aging management activities as necessary.
 - A public meeting to discuss the draft LR-ISG was held in late March 2012.
 - Final ISG in preparation.
- Comment [GL56]: Is this correct?
- Final LR-ISG-2011-05: Ongoing Review of Operating Experience
 - Issued March 16, 2012.
 - ADAMS Accession No. ML12044A215
 - Draft LR-ISG-2012-01: Wall Thinning Due to Erosion Mechanisms
 - Issued for public comment July 12, 2012; public comments by received August 27, 2012
 - ADAMS Accession No. ML12114A211
 - The LR-ISG describes the following:
 - o Allows wall thinning due to erosion mechanisms to be included in the "Flow-Accelerated Corrosion" aging management program as part of the "susceptible-not-modeled" category, and makes minor editorial changes.
 - o Revises the definitions of "wall thinning," "erosion," and "flow-accelerated corrosion," in the GALL Report to eliminate potential misinterpretations of these terms, and to align them with accepted industry definitions.
 - o Adds aging management review items to include erosion mechanisms in Engineered Safeguards, Auxiliary, and Steam and Power Conversion systems.
 - Socket Welds
 - Status of industry activities to develop non-visual examinations to ensure integrity of small-bore socket welds?
 - Metal Fatigue
 - A public meeting was held in early January 2012, in part to discuss methods to identify environmentally-assisted fatigue limiting locations.
 - NRC has reviewed an approach by Columbia, and is/will be evaluating approaches used by other applicants.
 - Containment Liner
 - NRC reports issued:
 - o ML112070867 - Containment Liner Corrosion Operating Experience Summary Technical Letter Report - Revision 1
 - o ML112150012 - Nuclear Containment Steel Liner Corrosion Workshop: Final Summary and Recommendation Report
 - Steam Generator Divider Plates and Tube-to-Tubesheet Welds
 - Foreign operating experience with cracking in Alloy 600 divider plates and/or 82/182 welds – concern with cracks extending to the pressure boundary.

- Interactions with Steam Generator Task Force to provide generic approaches to assure integrity of pressure boundary.
- Concern with cracking of tube-to-tubesheet welds with chromium content below that of Alloy 690 (consistency of once-through and recirculating steam generators).

IAEA International Generic Aging Lessons Learned (IGALL)

IAEA IGALL has three interaction levels – a Steering Group (SG) that provides overall direction for the program (John Lubinski, NRC Director of Division of License Renewal is the US member), a Clearing Group (CG) that ensures that the final products are consistent with SG direction, and three Working Groups (WGs) that are developing the final products. The three WGs (WG1 addresses Mechanical Components, WG2 addresses Electrical and I&C Components, and WG3 addresses Civil Structures and Components) are well along in completing their work; the CG has met multiple times to ensure common approaches and goals for each of the WGs. IGALL is using the basic approach in the US GALL report (NUREG-1801), although the aging management review (AMR) line item tables have a different arrangement from GALL and the aging management program (AMP) descriptions have nine elements. Time-limited aging analyses (TLAAs) are also included in the IGALL report, unlike in the NRC guidance documents where TLAAs are addressed in the License Renewal Standard Review Plan.

The remaining schedule for IGALL includes several meetings of the WGs, the CG and the SG, culminating in issuance of a final report and a Technical Meeting related to IGALL in September 2013. The WGs are meeting this spring to complete their work on AMR tables, AMP descriptions, TLAAs and other relevant parts of the IGALL report.

Technical and Regulatory Bases for Subsequent Renewal

NRC has initiated an activity to ensure adequate technical and regulatory bases for review of subsequent license renewal applications for operation to 80 years. Current activities include:

- Annual workshops to monitor industry/international technical progress
- Expand proactive materials degradation assessment to cover 80 years
- Collect/evaluate results from licensee implementation of license renewal aging management programs
- Continue and expand domestic and international partnerships

NRC held a public meeting on May 9, 2012, to receive public input on subsequent renewal, and has begun a series of webinars to solicit public input.

Research Activities

The NRC's Office of Nuclear Regulatory Research (RES) is undertaking several activities related to aging degradation and management research, including:

- **Expanded Materials Degradation Analysis (EMDA)** – In collaboration with and co-funded by the U.S. Department of Energy's Office of Nuclear Energy's (DOE:NE) Light Water Reactor Sustainability Program (LWRSP), the NRC has conducted an expert elicitation employing the Phenomena Identification and Ranking Technique (PIRT) with separate panels of experts for concrete and cable aging to identify the most significant aging degradation phenomena for which low and medium knowledge currently exists and which may need additional technical research. Two other panels on Reactor Pressure Vessel (RPV) and Core Internals and Piping are in the process of conducting the PIRT exercise. This work expands on the original NUREG-6923 (PMDA), and will provide technical insights on challenges for systems, structures, and components (SSC) for reactor operation beyond 40 years to at least 80 years of operating life. The several panels include a diverse body of experts representing regulatory bodies, industry (EPRI, vendors, etc.), U.S. national laboratories, academia, and international organizations. It is expected that, like EPRI's Materials Degradation Matrix (MDM) and associated Issues Management Tables (IMT), this effort will be repeated with some periodicity (perhaps every five years). The technical basis documents of the four volumes of the draft EMDA report is are under review by the panel and the staff. A NUREG/CR on EMDA is expected to be published by the end of C-Y-2013.
 - **Pilot Assessments of AMP Effectiveness at NPPs during Period of Extended Operation (PEO)** – NRC has conducted three AMP effectiveness audits at different reactor types (Westinghouse 2-loop PWR, Westinghouse 3-loop PWR, and GE BWR Mark 1 plant) to review how the AMPs in NUREG-1801 are implemented during the PEO and to gather information for potential revision of license renewal guidance documents (LRGDs) for subsequent license renewal (SLR). A technical letter report (TLR) is expected to be publicly available by the middle of C-Y-2013.
 - **Analysis of Periodic Safety Reviews (PSRs)** – NRC conducted a pilot review of selected translated PSR reports and related documentation from foreign nuclear regulatory authorities to identify any potential new regulatory insights regarding license-renewal-related topics and NPP operating experience. A technical letter report (TLR) is expected to be publicly available by the middle of C-Y-2013.
 - **International Forum on Reactor Aging Management (IFRAM)** – Monthly conference calls have been ongoing with the IFRAM Global Steering Committee, and a draft Charter, Operations Guidelines, a Proposal to Establish the IFRAM, and Desired Attributes have been developed and agreed upon in principle. The IFRAM members are now working on three tasks:
 - o Continuing development of best practices guideline document
 - o Pursuing development of International Agreement (IA) to fund and coordinate IFRAM activities
 - o FY-13 activities will focus on identifying aging management activities being conducted by participating organizations (and/or countries).
8. **Regulatory Issue Summary (RIS) 2012-12 – Small Modular Reactor (SMR) Designs**

On December 28, 2012, the NRC issued *Regulatory Issue Summary (RIS) 2012-12, "Licensing Submittal Information and Design Development Activities for Small Modular Reactor Designs."* The intent in issuing the RIS was to obtain new or updated information on the scheduling of a power reactor construction permit (CP), early site permit (ESP), combined operating license (COL), standard design certification (DC), standard design approval (DA), or manufacturing license (ML) application submissions related to SMR designs. The NRC anticipates receiving a number of CP, ESP, COL, DC, DA, and ML applications, starting as early as 2013, for a number of SMR designs. These designs include integral pressurized water reactors, high-temperature gas-cooled reactors, liquid-metal-cooled reactors, and other SMR designs. The purpose of the RIS was to help establish a predictable and consistent method for reviewing applications.

DC applicants form design-centered working groups (DCWGs) to facilitate the standardization of COL applications. The NRC staff also seeks information on potential DCWGs for each of the designs that may interact with the staff on generic or technology-related policy or technical issues. The NRC must identify possible applications and other interactions to formulate resource needs and budget requests for future fiscal years. The NRC encourages potential applicants to provide the agency with design and licensing plans, construction plans, and pre-application activities that will be used to demonstrate compliance with the NRC's safety and environmental requirements. In addition, information that potential applicants submit to the NRC will allow the agency to coordinate pre-application activities and, as appropriate, conduct vendor audits before the submission of applications. Furthermore, it will facilitate a more efficient licensing review of the applications. The RIS included a list of questions that respondents could use to provide the NRC with realistic, best-estimate predictions of applications or other submittals.

9. IN 2012-21 – RPV Head Studs

On December 10, 2012, the NRC issued Information Notice (IN) 2012-21, "Reactor Vessel Closure Head Studs Remain Detensioned During Plant Startup," to inform addressees of an event involving detensioned reactor vessel closure head studs at a boiling-water reactor that resulted in leakage from the reactor vessel during startup operations and a manual scram. Investigation activities determined that the reactor vessel head studs were not fully tensioned during startup operations and therefore, an unanalyzed condition existed. It was determined that none of the 64 reactor vessel head studs were adequately tensioned. The event was the result of errors made while operating the reactor vessel head stud tensioning equipment and during the validation process to ensure the head was properly tensioned. Specifically, the failure to provide proper training and lack of procedure guidance to correctly interpret critical data used to validate that the reactor vessel head studs are properly tensioned.

NRC Special Inspection Team Findings

An NRC special inspection team reviewed the licensee's actions prior to the event and identified examples of improper procedure adherence that contributed to the inadequate reactor vessel head stud tensioning. Specifically, the team determined that licensee personnel failed to

properly pressurize the reactor vessel head stud tensioning equipment to the value specified in Procedure OSMP-RPV502 because the tensioning equipment operators did not know how to correctly interpret the hydraulic pressure reading on the tensioning equipment display. The inspection team also determined that quality control personnel failed to verify proper reactor vessel stud elongation in accordance with stud elongation values specified in Procedure OSMP-RPV502. Further, the inspection team determined that nine of the twelve refuel floor personnel performing reactor vessel reassembly did not have the necessary refuel floor support training, as required by Procedure TRN-NGCC-1000, "Conduct of Training." Finally, based on its review of Procedure OPLP-20, "Post Maintenance Testing Program," which specifies "plant equipment shall be tested consistent with their safety functions following maintenance activities that may have impaired proper functioning of the components," the inspection team determined that the licensee failed to specify an adequate post maintenance test to verify the pressure retaining capability of the reactor vessel following a mid-cycle maintenance outage.

Section 50.120, "Training and qualification of nuclear power plant personnel," of 10 CFR states, in part, that the training program must incorporate the instructional requirements necessary to provide qualified personnel to operate and maintain the facility in a safe manner in all modes of operation. Criterion V, "Instructions, Procedures, and Drawings," of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50 states, in part, that instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.

This event also highlights the importance of human performance and oversight of maintenance activities. For example, operators of the stud tensioning equipment were not familiar with the pressure display, yet they proceeded with tensioning based on an incorrect interpretation of indicated tensioner pressure. In addition, a licensee lead mechanic and a quality control inspector signed a procedure checklist for stud elongation measurements using flawed data, based on incorrect explanations by other members of the maintenance crew. Other findings related to human performance can be found in the April 20, 2012, inspection report.

10. IN 2012-23 — Recent Radiography Events

On December 26, 2012, the NRC issued IN 2012-23, "Recent Radiography Events Resulting in Exposures Exceeding Regulatory Limits," to alert the industry of recent events that resulted in radiography workers receiving occupational doses in excess of the dose limits specified in 10 CFR 20.1201.

Over the past year, NRC has received four radiography event reports of radiography workers having received occupational whole body doses in excess of the 0.05 sievert (5 rem) limit or the 0.5 Sv (50 rem) to the skin of any extremity. The events are summarized in the IN. All events involved a QSA Global radiography camera, Model 880 Delta. NRC is highlighting these occurrences as a reminder to workers of the limits of personal dosimetry, even when properly working and worn. In instances when overexposure is possible and the dose on the dosimetry is questionable, licensees are encouraged to assess the situation to see if a re-enactment of the event would be prudent.

In the first event, a radiographer climbed a ladder to remove the source guide tube from the camera, and the source was not in the shielded position. The licensee calculated the dose to the radiographer's hands to be 58.15 cSv (rem). In the second event, a radiography crew performing operations on a pipeline project noticed that the locking mechanism on the camera had not popped up. The radiographers confirmed that their survey meters read zero. However, one radiographer's rate alarm was chirping, and the other radiographer's rate alarm was silent. Problems had been identified with both radiographers' rate alarms prior to beginning work, but operations were still conducted. The radiographers checked the crank assembly and were able to make approximately one turn to fully retract the source. Their personnel dosimeters were sent for emergency processing and results revealed whole body exposures of 5.133 and 1.447 cSv (rem). The third event involved a radiographer working in a shooting bay at a fixed facility. While carrying a dose-rate meter, the radiographer entered the shooting bay to set up for the next operation but was not paying attention to the dose-rate meter. The radiographer completed set-up, left the shooting bay, attempted to crank the source out, but then discovered that the source was already cranked into the collimator. The licensee determined that the radiographer received a TEDE of 8.1 cSv (rem), based on a total exposure time of two minutes and 30 seconds. The fourth event involved a radiographer trainer (RT) working on a scaffold. Thinking the source had been properly retracted; the RT disconnected the source guide tube from the camera and, with the guide tube around his neck, climbed down the scaffold ladder. A radiographer trainee then had problems disconnecting the crank assembly from the camera and the camera locking mechanism was still unlocked. Radiation surveys of the camera and guide tube revealed radiation levels indicating that the source was still within the guide tube. Both the RT's and the trainee's alarming rate meters sounded at some point during the survey. The RT picked up the guide tube with long tongs and the source fell onto the deck. The RT's film badge was sent for immediate processing, and the results revealed a deep-dose equivalent (DDE) whole body dose of 0.82 mSv (812 mrem). However, as the result of a re-enactment, the licensee calculated the dose at 29.32 cSv (rem). Blood tests were normal and no symptoms of local radiation injury were identified. The manufacturer examined the equipment and made the following determinations: (1) the drive cable was rusted, corroded, stiff, and lacked lubrication, leading to severing directly behind the 550 connector, (2) the male connector passed the no-go test, but was heavily worn, and (3) the control assembly components revealed significant signs of rusting and the housing was taped together to allow continued use. The manufacturer's overall conclusion was that the cable failed due to a combination of wear, corrosion, and lack of lubrication.

Comment [GL57]: Too much detail? Consider deleting and keeping the sentence in the preceding paragraph?

In most of these events, a functioning survey and/or rate meter was available, but was not properly utilized in a preventive capacity. Also, some type of inattention to detail was a factor in all of these events. Because of the high potential doses associated with radiography, licensees should always have calibrated, functioning survey meters that are used when approaching the radiography camera or guide tube after an exposure. Likewise, calibrated, functioning personal rate alarms should always be utilized. NRC understands that survey instruments and alarming rate meters can fail to work for a variety of reasons, or the meter may appear to be working, but may respond slowly to initially indicate a lower incorrect dose-rate reading. Because of this, and because of the serious dangers involved in using radiography sources, both instruments are required to be used during radiography. Radiographers are reminded to follow proper safety procedures when using radiography equipment. It should also be noted that in three of the four events discussed, re-enactments were necessary to estimate a more accurate dose although

dosimetry was worn. In fact, in one event, the dosimetry severely underestimated the radiographer's dose.

11. Doel 3

Four representatives of the NRC attended 1-day working group meetings held on October 16, 2012 hosted by the Federal Agency for Nuclear Control (FANC) in Brussels, Belgium. The purpose of this trip was to support a FANC request for NRC participation on three technical expert working groups in support of Belgian nuclear safety authority's assessment of the indications detected during examinations performed in June and July 2012 in the beltline shell region of the Doel Unit 3 reactor pressure vessel (RPV). Similar indications were also found later at Tihange Unit 2. The objectives of these working group meetings were:

- Share information and experience between nuclear safety authorities on regulatory approaches and actions in relation with the Doel 3 issue.
- Taking into account the lessons learned from the Doel 3 issue, discuss actions to be considered in other countries.
- Provide technical advice to Belgian nuclear safety authorities (FANC, Bel V, AIB Vinçotte) on specific topics / questions related to the Doel 3 & Tihange 2 RPV issue (note, however, that the actual evaluation of potential continued operation of the Doel 3 and Tihange 2 reactors remains the responsibility of the Belgian nuclear safety authorities).

The meeting consisted of three expert working groups composed of expert members of foreign nuclear safety authorities or related organizations (NRC, IAEA, NEA, JRC Petten, etc.) that were willing to participate in this issue. The three expert working groups were as follows:

- Expert Working Group 1 - Non-Destructive Examination Techniques
- Expert Working Group 2 - Metallurgical Origin / Root Causes of the Flaw Indications
- Expert Working Group 3 - Structural Mechanics and Fracture Mechanics – Approach for Justification

The chairman and technical secretary for each working group were provided by Belgian nuclear safety authorities.

FANC's desired outcome for the working group meetings was to provide support to the Belgian nuclear safety authorities for defining essential questions that the Belgian licensee must address to satisfy safe continued operation of the Doel 3 plant.

For the NRC, the desired outcome of attendance at the working group meetings was to obtain information such that the NRC can determine the importance of the flaws detected at Doel 3 and Tihange 2 that can be used to assess the importance of this issue on the U.S. fleet of reactors.

Additional working group meetings may be held prior to the end of 2012 in Belgium, and the NRC intends to schedule a public meeting with U.S. industry representatives sometime in December.

Belgian authorities have provided information summaries on their public website:
<http://www.fanc.fgov.be/page/homepage-federaal-agentschap-voor-nucleaire-controle-fanc/1.aspx>

Comment [GLS8]: Omitted – Bob requested to update all.