NRC Report
May 2011

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

Draft Final Rule – 2005 through 2008 Addenda

A proposed rule was published in the Federal Register on May 4, 2010 (75 FR 24324). The proposed amendment to 10 CFR 50.55a would incorporate by reference:

- The 2005 Addenda through 2008 Addenda of Section III, Division 1, and Section XI, Division 1, of the *Boiler and Pressure Vessel Code*;

- The 2005 Addenda and 2006 Addenda of the *Code for Operation and Maintenance of Nuclear Power Plants*;

- Code Case N-722-1, "Additional Examinations for PWR Pressure Retaining Welds in Class 1 Components Fabricated With Alloy 600/82/182 Materials Section XI, Division 1;" and

- 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 Weld Filler Material With or Without Application of Listed Mitigation Activities, Section XI, Division 1."

The rule would also:

- clarify which portions of Section III are approved for use by applicants and licensees
- identify which portions of Section III are NRC requirements, and which portions of Section III are not required to be implemented by 10 CFR 50.55a
- substitute the word "condition(s)" for the words "limitation(s)" "modification(s)" and "provision(s)" throughout 50.55a for consistency
- clarify the time frame for licensees to submit requests for relief based on impracticality for IST and ISI
- allow the use of 1994 Edition of NQA-1, "Quality Assurance Requirements for Nuclear Facility Applications," when using the 2006 Addenda of Section III of the ASME B&PV Code and later editions and addenda.

Finally, the NRC also requested comments on what the scope of the ASME B&PV Code edition and addenda rulemaking should be; how often the NRC should incorporate Code editions and addenda into 10 CFR 50.55a; and in what ways the NRC should communicate the scope, schedule for publishing the rulemakings in the Federal Register, and status of the 10 CFR 50.55a rulemakings to external users.

The public comment period closed on July 19, 2010, and the NRC received 22 public comment letters. The NRC has evaluated the comments. The final rule is scheduled for publication in June 2011. It should be noted that the NRC is considering approving Code Case N-770-1 in the final rule.

Restructuring of 10 CFR 50.55a / Public Workshop

The NRC has received feedback that the format of 10 CFR 50.55a can be confusing. In addition, questions have been raised regarding the clarity of certain requirements and the potential for inconsistent implementation. The NRC staff has held several recent internal meetings to discuss the potential for restructuring this section of the regulations and clarifying any confusing language.

The staff is currently developing an outline regarding a possible new format and a plan to address resource issues. Office of *Federal Register* requirements preclude having multiple proposed rules addressing the same provisions open at the same time. Therefore should the effort to modify 10 CFR 50.55a progress, the modification of the regulation must be incorporated with one of the routine updates to approve a new edition Any modification effort, therefore, has to be carefully considered to ensure that the endorsement of a new edition is not unduly delayed by the parallel modification effort (restructuring or clarity). Such modifications will have to be widely evaluated by all interested parties prior to inclusion in a rulemaking.

The staff believes that restructuring and clarification should be separate efforts accomplished in two phases. The first phase would be the restructuring of the regulation. Issues regarding clarity would be addressed in phase two. The staff is considering approaches such as public workshops and Advanced Notice of Public Rulemaking to ensure that any potential modifications have been widely evaluated prior to inclusion in a new edition rulemaking.

A technical basis substantiating any changes to 10 CFR 50.55a will have to be developed that addresses the benefits and burdens of proposed changes. For example, licensee procedures often reference particular paragraphs of the regulation. Restructuring of the regulation will create the burden of revising procedures to reference new paragraph numbers. It would be helpful to the staff in developing the technical basis if resource estimates were available for addressing this burden. Similarly, the staff is interested in the benefits of restructuring. Any input can be informally provided though email to Wallace Norris, NRC representative on ASME Technical Oversight Management Committee, at wallace.norris@nrc.gov.

2. ASME Code Case Rulemaking/Regulatory Guides

On October 5, 2010, Regulatory Guide 1.84, Revision 35, “Design, Fabrication, and Materials Code Case Acceptability, ASME Section III,” and Regulatory Guide 1.147, Revision 16, “Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1,” were issued as final guides (75 FR 61530). The guides address Code Cases from Supplement 2 to the 2004 Edition through Supplement 0 to the 2007 Edition (Supplement 0, 2007 Edition also serves as Supplement 12 to the 2004 Edition).

On October 5, 2010, Regulatory Guide 1.193, Revision 3, “ASME Code Cases Not Approved for Use,” was issued as a final guide (75 FR 61531).

On October 5, 2010, the final rule that incorporated Regulatory Guide 1.84, Revision 35, and Regulatory Guide 1.147, Revision 16, by reference into 10 CFR 50.55a was also issued (75 FR 61321).

Draft Revision 36 to RG 1.84, draft Revision 17 to RG 1.147, draft Revision 2 to RG 1.192, and draft Revision 4 to RG 1.193 are scheduled to be published for public comment in the July / August 2011 timeframe. The draft guides will address Supplements 1 – 10 to the 2007 Edition. The staff plans to address the issues raised by Raymond A. West in a petition for rulemaking dated December 14, 2007, and revised on December 19, 2007, in the July 2011 proposed ASME Code Case rulemaking. It should be cautioned, however, that the proposed rulemaking is still under review and thus subject to change.

The NRC staff has completed its review of Supplements 1 − 11 to the 2007 Edition.

3. Risk-Informed Activities

The Standards Committee on Nuclear Inservice Inspection (Section XI) has been considering a revision to Code Case N-716, “Alternative Piping Classification and Examination Requirements, Section XI, Division 1.” The NRC has been reviewing EPRI Topical Report 1018427, "Nondestructive Evaluation: PRA Technical Adequacy Guidance for RI-ISI Programs,” that was developed, in part, as a technical support document for the Code Case. Given that the topical report was still under review, the NRC staff representative to Section XI disapproved the proposed revision when it was considered by the committee.

The staff has completed its review of the topical report and is currently developing a report on its findings. The staff’s report will be available mid-2011. While any information relative to final positions on the topical are still considered preliminary, it can be reported that the NRC will request a small number of changes to the topical, and final approval will be conditional on those changes being incorporated into the revised topical.

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

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 proposed amendment to 10 CFR 50.55a (see #1 above). NRC staff are currently reviewing public comments on the proposed rule to address NRC conditions and the inclusion of ASME Code Case N-770-1 in lieu of N-770 in the final rule. Once the final rule is published, NRC staff will hold a public meeting to address the implementation of these new requirements. The public meeting date is to be determined (anticipated in July 2011).

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

5. New Reactor Licensing Activities

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.

New Reactor Licensing Status

As of April 15, 2011, the status of new reactors licensing under 10 CFR Part 52 is as follows:

Design Certification

NRC has issued four design certifications to date (ABWR, System 80+, AP-600, and AP-1000). These are certified in 10 CFR Part 52, Appendices A, B, C, and D, respectively. The NRC is currently reviewing four design certifications:

- General Electric-Hitachi's ESBWR (first passive BWR)
- AREVA's EPR (evolutionary pressurized-water reactor)
- Mitsubishi Heavy Industries' US-APWR (advanced pressurized water reactor)
- AP-1000 Revision 17 (first amended design certification)

On February 24, 2011, the NRC issued in the *Federal Register* a proposed rule (76 FR 10260) for comment on the AP1000 design certification amendment. The NRC is proposing to amend its regulations in 10 CFR Part 52, Appendix D to certify a revision to the AP1000 standard plant design that, in part, addresses the effects of the impact of a large commercial aircraft, incorporates design improvements, and increases standardization of the design. The applicant for this amendment is Westinghouse Electric Company, LLC. The public is invited to submit comments on this proposed design certification rule (DCR), the revised generic design control document that would be incorporated by reference into the DCR, and the environmental assessment for this amendment to the AP1000 design. The proposed rule is also available on the Federal eRulemaking Portal http://www.regulations.gov under Docket ID NRC-2010-0131. Comments on the proposed rule will be accepted through May 10, 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. The NRC is proposing to amend its regulations in 10 CFR Part 52 to certify the ESBWR standard plant design. The applicant for the certification of the ESBWR design is GE-Hitachi Energy (GEH). The proposed rule is also available on the Federal eRulemaking Portal http://www.regulations.gov under Docket ID NRC-2010-0135. Comments on the proposed rule will be accepted through June 7, 2011.

Early Site Permits (ESPs)

NRC has issued four ESPs to date (Clinton, Grand Gulf, North Anna, and Vogtle). The NRC's issuance of the Vogtle ESP on August 26, 2009, is the first to be 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. This is the first ESP application for a greenfield site with no specific technology established at this time.

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, like the ESP application for the Victoria County Station, uses the Plant Parameter Envelope (PPM) approach which means no reactor design has been selected.

Combined License (COL) Applications

The North Anna, Unit 3 COLA was revised on June 29, 2010, to change its standard plant design from an ESBWR to a US-APWR.

NRC is currently reviewing 12 COL applications (20 new reactor units):

- 1 ABWR South Texas Project 3 and 4
- 7 AP-1000 Vogtle 3&4, William S. Lee Station 1&2, Shearon Harris 2&3, V.C. Summer 2&3, Levy County 1&2, Bellefonte 3&4*, and Turkey Point 6&7
- 4 ESBWR Fermi 3, Grand Gulf 3*, River Bend 3*, Victoria County 1 and 2**
- 3 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

Advanced Reactors Program

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

- High Temperature Gas-Cooled Reactors:
  - Next Generation Nuclear Plant (DOE) –Pre-application interactions underway, DOE expected to announce the next phase in August/September 2011. NGNP will not meet its congressionally mandated deadline for completing construction of a demonstration unit by 2021.
- Small and Medium-size LWRs including Small Modular Reactors (SMRs) and Integrated PWRs (iPWRs):

- NuScale (iPWR) – NuScale Power is developing a modular, scalable 45 mWe iPWR, but financial problems challenging work activities has led to suspended operations. Design certification application expected mid-2012
- B&W mPower (iPWR)– B&W is developing a modular, scalable 125 mWe iPWR. Pre-application interactions underway; Design Certification application expected in late 2013/early 2014. Integrated test facility under construction. Testing expected to begin September 2011. TVA Clinch River is planning to build an mPower plant under the Part 50 licensing process. B&W will pursue a design certification in parallel when the design details are more finalized.
- Hyperion – pre-application activities initiated; no date announced
- Terra Power – no date announced

- Liquid Metal Fast Reactors:
  - GE-H PRISM – Prototype COL application expected as early as FY-2013
  - Toshiba 4S – Design Approval application expected as early as FY-2013

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 50.55(e) Programs for Reporting Defects and Noncompliance.

FY 11 Vendor Inspection Plans

- Commercial grade dedication organizations
- Manufacturing for valves (all new reactor Design Centers)
- AP1000 modular construction
- Manufacturing for steam generator tubes for AP1000
- Digital Instrumentation and Control for AP1000, USAPWR, and ABWR
- STP ABWR mechanical component fabrication
- AP1000 CRDMs
- AP1000 Engineering Design Verification

Vendor Inspection Reports completed, issued and planned inspections

- GE Hitachi Nuclear Energy, Wilmington, NC,– issued
- Toshiba, Yokohama and Fuchu, Japan (STP ABWR Digital I&C) – canceled

- Flowserve (Limitorque), Lynchburg, VA – issued
- SPX Copes Vulcan, Erie, PA – scheduled and postponed
- Cameron Measurement Systems, City of Industry, CA (NUPIC Observation) – completed
- Kinetrics, Toronto, Canada (NUPIC Observation) – completed
- CS Innovations, Scottsdale, AZ – completed
- Westinghouse Engineering Design Verification, Cranberry Township, PA - 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.

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. 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 will be 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 would ensure a consistent level of quality and safety in the design of pressure-boundary components such as the reactor vessel, piping, pumps, and valves and allow 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 has been working with standards development organizations (SDOs) from several countries (i.e., U.S., Japan, Korea, France, Canada and, recently, the Russian Federation) for the past 3 years to compare each countries' pressure-boundary Code requirements for Class 1, 2 and 3 vessels, piping, pumps and valves to the requirements of the ASME Boiler and Pressure Vessel Code, Section III. Similarities and differences are being documented in a database table and a results report. The Code-comparison effort is the first step to achieve harmonization of pressure-boundary codes and standards. A draft of the final Code-comparison report and tables has been completed for Class 1, 2 and 3 vessels, piping, pumps and valves for Korea, Japan, and France with Canada and Russia following shortly thereafter. A copy was provided to the ASME Board on Nuclear Codes and Standards in March 2011 for review and comments. The final report is expected to be issued by May 2011. An MDEP/CSWG meeting was held in Paris, France, on April 18-20, 2011 with the SDOs to discuss the final report and results.

The MDEP/CSWG recently issued letters to ASME, AFCEN (France), JSME (Japan), CSA (Canada) and the KEPIC (Korea) SDOs to (1) encourage completion of the Code Comparison project and (2) ask these SDOs to address their plans to preclude future divergences of Code requirements. The MDEP/CSWG will be sending a similar letter to the NIKIET (Russia) SDO.

B. 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.

On September 8 -10, 2010, and September 15 – 17, 2010, NRO staff observed the Korean Regulator perform vendor inspections at Westinghouse Newington Operations in Newington, NH, and Flowserve in Vernon, CA. On September 13 – 17, 2010, JNES observed an NRO inspection at IHI in Yokohama, Japan, and on November 15 – 19, NNSA observed an NRO inspection at Consolidated Power Supply in Birmingham, AL.

On December 1-3, 2010, NRO staff participated at the VICWG meeting in Paris, France. During this meeting, plans were finalized for conducting a joint pilot inspection by the NRC and KINS at a Korean vendor facility. This effort, which will be led by KINS, is tentatively scheduled for October 2011.

On May 11 - 13, 2011, NRO staff will participate at the next meeting of the VICWG in Paris, France.

6. LICENSE RENEWAL ACTIVITIES

Following are on-going activities related to license renewal:

Current status of applications, staff reviews and approvals

- 66 units approved (new plants are Vermont Yankee, Kewaunee, and Palo Verde 1/2/3)
- 11 applications (16 units) under review
  - 1 (1 unit) awaiting final approval (Pilgrim)
  - 4 (7 units) completed ACRS full committee (Indian Point 2 & 3, Prairie Island 1 & 2, Hope Creek and Salem 1 & 2)
  - 2 (3 units) awaiting ACRS full committee (Crystal River 3 and Diablo Canyon 1 & 2 [7/2011]) – schedule for renewed license issuance for Crystal River 3 is impacted by containment concrete issues; Diablo Canyon has requested that renewed license issuance be delayed pending 3-D seismic studies, which is expected prior to December 2015
  - 4 (5 units) awaiting ACRS subcommittee (Seabrook [9/2011], Columbia [10/2011], Davis-Besse [3/2012], and South Texas Project 1 & 2 [8/2012])
- 3 applications with scheduled application dates through 2011

- June 2011 – Limerick 1 & 2
- July 2011 – Grand Gulf
- October to December 2011 – Callaway
- Others staggered out to 2017

Nine plants have entered the operating period beyond 40 years:

- Oyster Creek – April 9, 2009
- Nine Mile Point Unit 1 – August 22, 2009
- Ginna – September 19, 2009
- Dresden Unit 2 – December 22, 2009
- H.B. Robinson – July 31, 2010
- Monticello – September 8, 2010
- Point Beach Unit 1 – October 5, 2010
- Dresden Unit 3 – January 12, 2011
- Palisades – March 24, 2011

Technical Issues

Recent reviews of license renewal applications and plant operating experience have identified issues in several areas, as described below. These issues have resulted in changes to the Revision 2 of the GALL Report (NUREG-1801), which was issued in December 2010, and discussions with applicants with on-going reviews.

The NRC is considering a Regulatory Issue Summary (RIS) to notify licensees of the technical changes made in Revision 2 of the GALL Report such that they can assess the impact of these changes and the underlying bases for the changes on their plants.

- PWR Reactor Vessel Internals
  - NRC has issued a draft safety evaluation (SE) on MRP-227 report; expect to finalize the SE in June. Industry will then issue MRP-227-A as an approved report with conditions and limitations, and action items.
  - NRC is preparing a Regulatory Issue Summary (RIS) to describe implementation of MRP-227-A by applicants and licensees.
  - NRC is considering a License Renewal Interim Staff Guidance (LR-ISG) to implement changes to GALL Revision 2 aging management review line items and aging management activities as necessary.
- Buried Piping
  - NRC is considering an LR-ISG to address:
    - Plants without cathodic protection – soil sampling and a larger number of excavated direct visual inspections, depending on soil sampling results, and plant specific OE has found issues (e.g., coating degradation, leaks, unacceptable materials in backfill).
    - Inspection sample size for UT inspections of underground piping.
    - Provide further clarity on backfill specifications.
- Socket Welds
  - Status of industry activities to develop non-visual examinations to ensure integrity of small-bore socket welds?
- Metal Fatigue

  - Discussions with industry on how to demonstrate that environmentally-assisted fatigue has been considered for the limiting reactor coolant system components.
- Containment Liner
  - NRC is completing an activity to review operating experience and assess likelihood of corrosion occurrence – reports should be available soon.
- 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 (Brian Holian, 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 Electrical and I&C Components, and WG3 Civil Structures and Components) have begun their work; the CG had its first meeting 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.

The schedule for IGALL includes a series of WG, CG and SG meetings, culminating in issuance of a final report and a Technical Meeting related to IGALL at the end of 2013.

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

Several recent meetings considered both first and second renewals:

- The “Second Workshop on U.S. Nuclear Power Plant Life Extension Research and Development,” February 22 – 24, 2011, L’Enfant Plaza Hotel, Washington, D.C. (http://www.prc8a.com/LB60Workshop2011/)

– A Commission “Briefing on Reactor Materials Aging Management Issues,” March 1, 2011 (information can be linked from http://www.nrc.gov/reading-rm/doc-collections/commission/tr/2011/).

– A Regulatory Information Conference double session on materials aging and license renewal on March 9, 2011 (http://www.nrc.gov/public-involve/conference-symposia/ric/).

NRC is considering several public meetings starting in fall 2011 to receive public input on subsequent renewal. Several of these meetings may be held in conjunction with standards and codes development organizations, such as ASME Code.

Research Activities

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

- Expanded Proactive Materials Degradation Analysis (On-going) – In collaboration (and co-funding) with the U.S. Department of Energy’s Office of Nuclear Energy’s (DOE:NE) Light Water Reactor Sustainability Program (LWRSP), the NRC has initiated an expert panel to utilize a modified Phenomena Identification and Ranking Technique (PIRT) process to evaluate aging degradation phenomena on the Core Internals and Primary Systems; Balance of Plant Systems and Components; Reactor Pressure Vessel (RPV); Concrete Structures; and Electric Cabling. This work expands on the original NUREG 6923 (PMDA), and looks out 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), that this effort will be repeated with some periodicity (perhaps every five years). The time line for completing this effort is December 2011.

- Second Workshop on U.S. Nuclear Power Plant Life Extension Research and Development – The NRC, DOE:NE, and the Nuclear Energy Institute (NEI) co-sponsored a "Second Workshop on U.S. Nuclear Power Plant Life Extension Research and Development," February 22-24, 2011. The initial Workshop, co-sponsored by NRC and DOE in February 2008, focused on the technical issues that may need additional research efforts to support safe long-term operations (LTO) in the second, and subsequent, license renewal period (summary provided at ADAMS Accession Number ML080570419). This Workshop reviewed the accomplishments to date, discussed ongoing industry LTO activities, and discussed the remaining gaps that may need additional research. Based on the results of this workshop, the NRC staff concluded that significant additional progress needs to be made to resolve technical issues identified.

- IAEA Third Plant Life Management (PLiM) Symposium – The International Atomic Energy Agency (IAEA) has asked the U.S. to host the Third Plant Life Management (PLiM) Symposium in the Summer of 2012. Both DOE and NRC have agreed to co-host this meeting, and it will be held in Salt Lake City, Utah, 14-18 May 2012, with a side trip to the Idaho National Laboratory for interested parties. Following the initial planning meeting at IAEA headquarters in Vienna, Austria (February 9-10, 2011), a U.S. Steering

Committee is being formed, with governmental, industry, academia, and non-governmental organizations invited to participate (and co-sponsor) this meeting.

- International Forum on Reactor Aging Management – 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. A proposed kick-off meeting is being scheduled to coincide with the "15th International Conference on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors" in Colorado Springs (August 7-10, 2011), with the aim of instituting collaborative research activities among the several IFRAM members.

7. Buried Piping

Recent leaks from buried piping at nuclear power plants have caused the NRC to undertake a focused look at how underground piping is designed, maintained, and inspected to ensure structural integrity and to prevent leaks that could harm the environment. These leaks generated significant stakeholder interest, including inquiries from several congressmen. On December 2, 2009, the NRC staff responded to the Chairman's memorandum dated September 3, 2009, ADAMS No. ML092460648, tasking the staff to describe the activities currently underway or planned addressing the issue of leaks from buried piping. The response is SECY-09-0174, "Staff Progress in Evaluation of Buried Piping at Nuclear Reactor Facilities," and can be found at http://www.nrc.gov/reading-rm/doc-collections/commission/secys/2009/. Actions described in the SECY are being tracked in a buried piping action plan (ADAMS No. ML102590171).

A new Code Case N-XXX is under development to address underground piping systems. The staff has identified several areas that are not yet addressed, or require significant additional detail. The staff has provided its comments to Section III.

With regard to the integrity of buried piping systems and the prevention of groundwater contamination, the staff has initiated discussions with cognizant ASME committees regarding three issues. The first issue is that the Section III Code Cases currently under development do not address the ASME Code requirement for the design and arrangement of system components to allow for adequate access and clearances to conduct examination and tests. Lack of access for the inspection of buried and underground safety-related Class 3 piping has been identified as a serious issue relative to ensuring piping integrity. The second issue is the need to re-examine the scope of buried and underground piping in the ASME Code. Presently, the ASME Code only addresses safety-related Class 3. Leakage of contaminated fluids from other buried piping systems has been reported. Leakage from these systems can affect system operation and may have radiological impacts. Accordingly, the staff submitted an Issue Sheet to the Subgroup on Industry Experience for New Plants for consideration. The staff proposed that Section III consider the development of requirements addressing accessibility for inspection of buried and underground piping. Systems to be included would be based on considerations such as function and consequence. The staff also proposed that Section XI consider the development of requirements addressing the inspection of buried and underground piping for new plants. The inspection of buried and underground piping at operating plants is the third issue. The staff is concerned that: ASME Code required testing and surveillance requirements for Class 3 buried and underground piping do not appear to be sufficient to identify corrosion, degradation and leakage; and leaks from other buried piping systems carrying tritium were discovered through voluntary licensee monitoring for radioactive tritium in groundwater monitoring wells rather than through inspection, testing, or monitoring of the piping.

With regard to Class 3 buried and underground piping, the Class 3 buried piping pressure boundary has degraded and become compromised at several plants. In some cases, the degradation was significant but had not yet challenged structural integrity. Current ASME Code requirements are not sufficient to identify either degradation or a leak. If left undiscovered, degradation of buried and underground Class 3 piping could progress to a point that structural integrity is threatened, particularly for piping that experiences general coating failure followed by general corrosion. If such a system contains tritium, groundwater monitoring would indicate the presence of a leak. For those piping systems that do not contain tritium, however, neither groundwater monitoring nor inspection or Code-required testing would identify degradation, and the piping could continue to deteriorate until there is loss of function.

With regard to buried and underground piping other than Class 3, industry assessments have shown that it is important to maintain the integrity of these systems as they can affect the reliability of plant operation and can have radiological and environmental impacts. The industry has initiated a number of activities. Section XI should assess these activities with regard to the need for inspection and testing.

With regard to Code Case N-755 regarding the use of high density polyethylene piping for underground systems, the staff has identified issues to Section III related to design life, joining, and non-destructive examination that will need to be addressed for the staff to endorse the Code Case.

8. Information Notice 2011-04

On February 23, 2011, the NRC issued Information Notice (IN) 2011-04, “Contaminants and Stagnant Conditions Affecting Stress Corrosion Cracking in Stainless Steel Piping in Pressurized Water Reactors.” The NRC issued this IN to inform licensees of the effects of contaminants and stagnant conditions on the potential for stress corrosion cracking (SCC) in stainless steel piping in PWRs. The IN is available in ADAMS at ML103410363.

Several events led to the issuance of the IN. In September 2008, personnel at Callaway detected a small leak from a through-wall flaw in the 2-inch diameter, Schedule 160, Class 2, Type 304 stainless steel pressurizer auxiliary spray pipe. The flaw was axially oriented and was located beneath a pipe support clamp. Subsequently, the licensee detected a second pipe support clamp with corrosion on the same piping. The failure analysis showed that the flaws were the result of transgranular stress corrosion cracking (TGSCC) originating at the outside surface. Pitting was also observed which is caused by exposure to chlorides. Chemical analysis of the pipe surface confirmed the presence of chlorides on the outside surface of the pipe.

The pressurizer auxiliary spray line is used during normal shutdown operation. During normal plant operation, the coolant in the line is stagnant, but pressurized by the normal charging system. A control valve in the line is normally closed and the line is insulated. The temperature of the line is the surrounding ambient temperature during normal operation. During periods of warm weather, the moisture content in the containment building is elevated. The licensee believes that, under these circumstances, it is likely that condensation will form in the crevice between the pipe clamp and pipe outside surface.

In October 2009, as a result of the Callaway findings, Wolf Creek Generating Station (Wolf Creek) personnel conducted liquid-penetrant testing on similar piping and detected several axial indications beneath the pipe support clamps of the pressurizer auxiliary spray line. On one of the clamps, a small quantity of boron crystals was observed, indicating a through-wall flaw in the pipe. The through-wall flaw was not located near any pipe welds. Based on a review of relevant operating experience, the licensee concluded that the outside diameter-initiated SCC (ODSCC) was most likely TGSCC due to the presence of chlorides.

From 2009 to 2010, personnel at San Onofre Nuclear Generation Station (SONGS) detected three leaks at Unit 2 and five leaks at Unit 3 in various Class 2 stainless steel pipes. The affected systems include refueling water storage tank (RWST) gravity feed line to charging pump, RWST line to the emergency core cooling system (ECCS) suction line, ECCS minimum return line to the RWST, and containment emergency sump to the charging pump.

The operating experience summarized above shows that, as nuclear plants age, SCC can potentially become an emergent degradation mechanism in PWRs for environments that contain chlorides or stagnant flow conditions. Licensees should be aware of the potential for SCC to occur in stainless steel in PWR applications. There have been cases of intergranular stress corrosion cracking (IGSCC) of austenitic stainless steels in PWRs. NRC IN 91-05, "Intergranular Stress Corrosion Cracking in Pressurized Water Reactor Safety Injection Accumulator Nozzles" and IN 97-19, "Safety Injection System Weld Flaw at Sequoyah Nuclear Power Plant, Unit 2," discuss cases of IGSCC in PWRs due to furnace sensitized materials being exposed to an oxygenated environment. IGSCC in austenitic stainless steels is usually attributed to sensitization of the material by excessive exposure of the material to temperatures between approximately 800 – 1,500°F, usually during fabrication. NRC IN 85-34 discusses the impact of heat tracing on a pipe and chlorides in the water inside the pipe that led to SCC at a nuclear plant.

Other related generic communications are:

- NRC Circular 76-06, "Stress Corrosion Cracks in Stagnant Low Pressure Stainless Piping Containing Boric Acid Solution at PWR's"
- Bulletin 79-17, "Pipe Cracks in Stagnant Borated Water Systems at PWR Plants"
- IN 79-19, "Pipe Cracks in Stagnant Borated Water systems at PWR Plants"
- IN 85-34, "Heat Tracing Contributes to Corrosion Failure of Stainless Steel Piping"

SCC can be managed effectively to minimize the potential for catastrophic pipe failure through stainless steel piping cleanliness control and limiting the contact with fluids (including sweat from personnel) or condensation that may contain halogens (chlorides and fluorides). Water chemistry can be used to minimize the adverse effect of oxygen and chloride on SCC. When welding piping joints or attachments, appropriate procedures can be followed to minimize stainless steel sensitization. Periodic inspections of the susceptible piping systems as part of the existing boric acid corrosion control program per the April 2008 Nuclear Energy Institute report NEI 03-08, Revision 1, "Guideline for the Management of Materials Issues," or as part of routine walkdowns have been instrumental in detecting SCC in stainless steel piping. By letter dated October 14, 2010, the PWR Owners Group issued PA-MSC-0474, "Outside Diameter Initiated Stress Corrosion Cracking Revised Final White Paper," which provides additional information on SCC.

9. Draft Regulatory Guide Related to Subsection IWL

On April 28, 2011, draft regulatory guide (DG)-1197, entitled, ''Inservice Inspection of Prestressed Concrete Containment Structures with Grouted Tendons,'' was published for public comment. The public comment period closes on June 26, 2011. DG–1197, proposed Revision 2 of Regulatory Guide 1.90, describes an approach that the staff of the NRC considers acceptable for use in developing an appropriate surveillance program for prestressed concrete containment structures with grouted tendons. The purpose of this guide is to provide recommendations for inservice inspection of containments and quality standards that should be maintained when Portland cement grout is used for the corrosion protection of prestressing steel. To view the draft guide, visit: http://www.nrc.gov/reading-rm/doc-collections/reg-guides/power-reactors/draft-index.html
The FRN for DG-1197 has been posted at http://www.gpo.gov/fdsys/pkg/FR-2011-04-28/pdf/2011-10336.pdf

10. Updated Regulatory Guides Addressing Welding

On March 15, 2011, the NRC issued Revision 1 to Regulatory Guide (RG) 1.34, "Control of Electroslag Weld Properties" (76 FR 14108). This revised RG describes methods that the NRC staff considers acceptable for implementing requirements about the control of weld properties when fabricating electroslag welds for nuclear components made of ferritic or austenitic materials. This RG applies to light-water reactors.

On March 15, 2011, the NRC issued Revision 1 to RG 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Components" (76 FR 14107). This revised RG describes a method that the NRC staff considers acceptable for the selection and control of welding processes used for cladding ferritic steel components with austenitic stainless steel to restrict practices that could result in underclad cracking. This guide is limited to forgings and plate material and does not apply to other product forms such as castings and pipe. Adequate resistance to underclad cracking for these latter items should be ensured on a case-by-case basis. This RG applies to light-water reactors.

On April 1, 2011, the NRC issued Revision 1 to RG 1.44, "Control of the Processing and Use of Stainless Steel" (76 FR 18262). This revised RG describes a method that the NRC staff considers acceptable for implementing requirements about control of the application and processing of stainless steel to avoid severe sensitization that could lead to stress-corrosion cracking. This RG applies to light-water reactors.

On March 15, 2011, the NRC issued Revision 1 to RG 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel" (76 FR 14108). This revised RG describes a method that the NRC staff considers acceptable for implementing regulatory requirements related to the control of welding for low alloy steel components during initial fabrication. This RG applies to light-water reactors.

All four of the above revised RGs are available through the NRC's public web site under '"Regulatory Guides" at http://www.nrc.gov/reading-rm/doc-collections/.