7
IMPLEMENTATION REQUIREMENTS

The purpose of this section is to summarize the implementation requirements of these guidelines. These guidelines do not reduce, alter, or otherwise affect current ASME B&PV Code Section XI or plant-specific licensing inservice inspection requirements.

7.1 NEI 03-08 Implementation Protocol

These guidelines are a 'work product' of the EPRI MRP, an 'Issue Program (IP)' as defined in NEI 03-08 [1]. Appendix B to NEI 03-08, Implementation Protocol, defines the processes and expectations for implementing industry guidance issued under the Materials Initiative, and requires that IPs identify the specific implementation category for 'requirements' identified by guideline-type work products.

The three implementation categories described in NEI 03-08 are as follows:

- Mandatory – to be implemented at all plants where applicable;
- Needed – to be implemented wherever possible, but alternative approaches are acceptable; and
- Good Practice – implementation is expected to provide significant operational and reliability benefits, but the extent of use is at the discretion of the individual utility.

Sections 7.2 through 7.7 list or summarize the requirements contained in this document. A failure to meet a Needed or a Mandatory requirement is a deviation from the guidelines and a written justification for the deviation must be prepared and approved as described in Appendix B to NEI 03-08 [1]. A copy of the deviation is sent to the MRP so that improvements to the guidelines can be developed.

7.2 Aging Management Program Requirement

Mandatory: *Each commercial U.S. PWR unit shall develop and document a program for management of aging of reactor internal components within thirty-six months following issuance of MRP-227-Rev. 0 (that is, no later than December 31, 2011).*

MRP-227-Rev. 0 is the first published version of these guidelines.

7.3 Reactor Internals Guidelines Implementation Requirement

Needed: *Each commercial U.S. PWR unit shall implement Tables 4-1 through 4-9 and Tables 5-1 through 5-3 for the applicable design within twenty-four months following issuance of MRP-227-A.*

Implementation of these guidelines is to take effect 24 months following issuance of MRP-227-A (that is, no later than December 31, 2013). Implementation means performance of inspections of applicable components within the time frame specified in the guidance provided in the applicable tables. MRP-227-A is the current version that has incorporated the changes proposed by the MRP in response to U.S. Nuclear Regulatory Commission (NRC) Requests for Additional Information, recommendations in the NRC Safety Evaluation and other necessary revisions identified since the previous publication of the report (MRP-227 Rev. 0).

Earlier implementation may be required by plant-specific regulatory commitments (for example, license renewal approvals). Plants implementing these guidelines prior to the issuance of the "NRC-approved" version would thus implement the requirements in accordance with the current published version of these guidelines.

Consistent with the requirements of NEI 03-08, if the guidance contained in Table 4-1 through 4-9 and/or Tables 5-1 through 5-3 cannot, need not, or will not be implemented as written, a technical justification must be prepared that clearly states what requirement cannot, need not, or will not be met and why; what alternative action is being taken to satisfy the objective or intent of the guidance; and, why the alternative action is acceptable. Examples of alternatives that may be justifiable are: elevation of an Expansion component to Primary; substitution of an equivalent or more rigorous examination than is required by the tables; or destructive testing in lieu of nondestructive examination, such as the case where one or more of the primary components is being replaced. Since the Expansion components are also "needed" requirements, the technical justification for not fully implementing a Primary component examination or not implementing it in a manner consistent with its intent, would be expected to include disposition of the associated Expansion components.

When submittal of a deviation from work products or elements is required, the justification shall be reviewed and approved in accordance with the applicable plant procedures with the additional responsibility for deviation from a 'Needed' element that an internal independent review is performed and that concurrence is obtained from the responsible utility executive. Further, as stipulated in the Implementation Protocol (Appendix B) of NEI 03-08, a utility is required to notify the Issue Program (e.g., the MRP) and the NRC.

7.4 Examination Procedures Requirement

Needed: *Examinations specified in these guidelines shall be conducted in accordance with the Inspection Standard [3].*

7.5 Examination Results Requirement

Needed: *Examination results that do not meet the examination acceptance criteria defined in Section 5 of these guidelines shall be recorded and entered in the plant corrective action program and dispositioned.*

7.6 Aging Management Program Results Requirement

Needed: Each commercial U.S. PWR unit shall provide a summary report of all inspections and monitoring, items requiring evaluation, and new repairs to the MRP Program Manager within

120 days of the completion of an outage during which PWR internals within the scope of MRP-227 are examined.

This summary of the results will be compiled into an overall industry report which will track industry progress, aid in evaluation of significant issues, identification of fleet trends and determination of any needed revisions to these guidelines. The industry report will be updated biennially for the benefit of the fleet, the regulator, the PWROG and other industry stakeholders. This biennial report will serve to assist in review of operating experience, and required monitoring and trending for aging management programs established by the industry. In order to ensure completeness and consistency of reporting, the MRP will provide a template listing the requested information.

7.7 Evaluation Requirement

Needed: If an engineering evaluation is used to disposition an examination result that does not meet the examination acceptance criteria in Section 5, this engineering evaluation shall be conducted in accordance with a NRC-approved evaluation methodology.

8
REFERENCES

1. *Guidelines for the Management of Materials Issues*, NEI 03-08, Nuclear Energy Institute, Washington, DC, Latest Edition.
2. *ASME Boiler & Pressure Vessel Code, Section XI, Division 1, "Rules for Inservice Inspection of Nuclear Power Plant Components,"* American Society of Mechanical Engineers, New York, NY, 2001 Edition, Plus 2003 Addenda, or later.
3. *Materials Reliability Program: Inspection Standard for Reactor Internals (MRP-228)*. EPRI, Palo Alto, CA: 2009. 1016609.
4. 10 CFR 50.55a – Codes and Standards, Title 10 (Energy), Part 50 (Domestic Licensing of Production and Utilization Facilities) of the Code of Federal Regulations, U.S. Nuclear Regulatory Commission, Washington, DC, 2005.
5. *Materials Reliability Program: Framework and Strategies for Managing Aging Effects in Reactor Internals (MRP-134)*. EPRI, Palo Alto, CA: 2005. 1008203.
6. *Materials Reliability Program: Characterization of Decommissioned PWR Vessel Internals Material Samples – Material Certification, Fluence, and Temperature (MRP-128)*. EPRI, Palo Alto, CA: 2003. 1008202.
7. *Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values (MRP-175)*. EPRI, Palo Alto, CA: 2005. 1012081.
8. *Materials Reliability Program: Screening, Categorization, and Ranking of B&W-Designed PWR Internals (MRP-189-Rev. 1)*. EPRI, Palo Alto, CA: 2009. 1018292.
9. *Materials Reliability Program: Failure Modes, Effects, and Criticality Analysis of B&W-Designed PWR Internals (MRP-190)*. EPRI, Palo Alto, CA: 2006. 1013233.
10. *Materials Reliability Program: Screening, Categorization and Ranking of Reactor Internals of Westinghouse and Combustion Engineering PWR Designs (MRP-191)*. EPRI, Palo Alto, CA: 2006. 1013234.
11. *Materials Reliability Program: Functionality Analysis for B&W Representative PWR Internals (MRP-229)*. EPRI, Palo Alto, CA: 2008. 1016598.
12. *Materials Reliability Program: Functionality Analysis for Westinghouse and Combustion Engineering Representative PWR Internals (MRP-230)*. EPRI, Palo Alto, CA: 2008. 1016597.
13. *Materials Reliability Program: Aging Management Strategies for B&W PWR Internals (MRP-231)*. EPRI, Palo Alto, CA: 2008. 1016592.
14. *Materials Reliability Program: Aging Management Strategies for Westinghouse and Combustion Engineering PWR Internals (MRP-232)*. EPRI, Palo Alto, CA: 2008. 1016593.

15. Letter to Reactor Internals Focus Group from MRP, Subject: *Minutes of the Expert Panel Meetings on Expansion Criteria for Reactor Internals I&E Guidelines*, MRP 2008-036 (via email), June 12, 2008.

16. *ASME Boiler & Pressure Vessel Code, Section V, Nondestructive Examination*, American Society of Mechanical Engineers, New York, NY, 2004 Edition, July 1, 2004.

17. *Nondestructive Evaluation: Evaluation of Remote Visual Examination Methods*. EPRI, Palo Alto, CA: 2006. 1013537.

18. Topical Report BAW-2248A, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals," March 2000.

19. *BWRVIP-100-A: BWR Vessel and Internals Project, Updated Assessment of the Fracture Toughness of Irradiated Stainless Steel for BWR Core Shrouds.* EPRI, Palo Alto, CA: 2006. 1013396.

20. *ASME Boiler & Pressure Vessel Code, Section XI, Division 1, Nonmandatory Appendices, Appendix C, "Evaluation of Flaws in Austenitic Piping*," American Society of Mechanical Engineers, New York, NY, 2004 Edition, July 1, 2004.

21. *ASME Boiler & Pressure Vessel Code, Section III, Division 1, Nonmandatory Appendices, Appendix F, "Rules for Evaluation of Service Loadings With Level D Service Limits*," American Society of Mechanical Engineers, New York, NY, 2004 Edition, July 1, 2004.

22. *Materials Reliability Program: Fracture Toughness Evaluation of Highly Irradiated PWR Stainless Steel Internal Components (MRP-210)*. EPRI, Palo Alto, CA: 2007. 1016106.

23. *BWR Vessel and Internals Project, Evaluation of Crack Growth in BWR Stainless Steel Reactor Pressure Vessel Internals (BWRVIP-14)*. EPRI TR-105873, EPRI, Palo Alto, CA: 1996.

24. *BWRVIP-99: BWR Vessels and Internals Project, Crack Growth Rates in Irradiated Stainless Steels in BWR Internal Components*. EPRI TR-1003018, EPRI, Palo Alto, CA: 2001.

25. *Materials Reliability Program: PWR Internals Age-Related Material Properties, Degradation Mechanisms, Models, and Basis Data – State of Knowledge (MRP-211)*. EPRI, Palo Alto, CA: 2007. 1015013.

26. WCAP-17096-NP, "Reactor Internals Acceptance Criteria Methodology and Data Requirements" Revision 2 (December 2009), or latest NRC-approved revision.

27. U. S. Nuclear Regulatory Commission letter "Revision 1 to the Final Safety Evaluation of EPRI Report, Material Reliability Program Report 1016596 (MRP-227), Revision 0, *Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines*, (TAC NO. ME0680)," dated December 16, 2011.

A
REACTOR INTERNALS OPERATIONAL EXPERIENCE

Note that in Revision 0 to MRP-227, Appendix A provided guidance for development of an Aging Management Program (AMP) for PWR internals components. This guidance has been deleted from MPR-227. Guidance for AMP preparation may be found in AMP XI.M16A of NUREG-1801, Revision 2 (or subsequent revisions).

Commercial PWR vessel internals in the United States have experienced safe, relatively trouble-free operation. There have been no instances to date in which PWRs in the U.S. have posed a threat to public safety as a result of PWR internals material aging degradation. While relatively few incidents of PWR vessel internals aging degradation have been reported in operating U.S. commercial PWR plants, a summary of the current operating experience is useful for licensees developing aging management programs. This summary is organized first by the aging effect and subsequently by the age-related degradation mechanism leading to that effect. This compilation neither replaces efforts by licensees to review and document their plant-specific operating experience that may impact plant programs, nor does it preclude licensee participation in industry initiatives that perform these functions.

Cracking

IGSCC — Multiple PWR internals bolt failures of the lower thermal shield bolts were discovered during the 1981 and 1982 in-service inspections performed at three B&W-design PWRs. The thermal shield bolt locking clips at these three plants were visually observed to be missing or loose. Subsequent examinations during 1982, 1983, and 1984 revealed bolt failures at four additional units. These failures included upper core barrel, lower core barrel, upper thermal shield, and surveillance specimen holder tube bolts. All of the affected fasteners were fabricated from Alloy A-286 ASTM A 453, Grade 660, Condition A or B material. The results of an extensive evaluation program revealed the failure mechanism was predominantly due to an environmentally-assisted IGSCC mechanism. However, for some bolts, there was evidence that fatigue was also a contributor, likely in the form of corrosion fatigue.

In general, the primary mechanism causing cracking and failure of the Alloy A-286 PWR internals bolts was IGSCC. All the failures occurred in the bolt head-to-shank fillet. Information Notice (IN) 90-68 provides information about IGSCC cracking in Alloy A-286 bolts used to hold the turning vanes to reactor coolant pumps at a foreign plant. The IN 90-68 document includes a general discussion of the problems experienced with cracking of Alloy A-286 bolting materials, including the problems identified with respect to B&W PWR internals bolting.

In 2005, cracking of replacement core barrel-to-former plate bolts fabricated from cold-worked Type 316Ti stainless steel was observed in a German PWR by visual inspection. These bolts had replaced the original Alloy X-750 core barrel bolts in the late 1980s, which had exhibited failure due to PWSCC (described below). Subsequent UT inspection and failure analysis confirmed that

the cracking was confined to the bolt head initiating from the bolt fillet transition, but the bolt threads and shank were free from cracking. The failure mechanism of the cold-worked Type 316Ti stainless steel replacement core barrel bolts has been identified as IGSCC. To date, all known failures of core barrel bolts have been limited to the original Alloy X-750 and the replacement cold-worked Type 316Ti stainless steel in German PWRs.

PWSCC — Alloy X-750 has experienced numerous worldwide failures in the Westinghouse-designed PWR internals involving the control rod guide tube support pins (a.k.a., split pins). As noted in IN 82-29, these failures first appeared in Japan in the late 1970s. Split pin failures prompted investigations and modifications to manufacturing practices. The original heat treatment condition AH[1] of the age-hardenable material has shown the most susceptibility to PWSCC cracking. By the early 1980s, nearly all of the original design split pins had been replaced with the improved HTH heat treatment Condition.

In 1987, failures of Alloy X-750 HTH Condition control rod guide tube support pins in French PWRs occurred at much shorter times and lower stresses than expected. Foucault, et al., showed that these early failures were due to the surface condition of the pins. Any heat treatment after machining degrades the performance of Alloy X-750. The greatest resistance to IGSCC was found when machining or polishing was performed after heat treatment, which removes an oxide layer from the surface of the material. Additional refinements have since been made to the manufacturing practices used to produce a newer version of Alloy X-750 HTH split pins.

After an extensive worldwide industry program to develop a material heat treatment for Alloy X-750 that would have maximum resistance to SCC, Westinghouse and utility customers conducted a campaign during the 1980s to replace guide tube support pins. Ultimately, Westinghouse developed a cold-worked Type 316 stainless steel support pin as a replacement and a number of utilities have performed replacements with this design. A few utilities have opted to perform ultrasonic inspections rather than initiate wholesale replacements, while still other utilities have preferred to take no action at this time.

Alloy X-750, in a condition similar to AH, was used for the baffle-to-former plate bolts in the German Biblis-type reactors. After about four years of service, several bolts were found either cracked or severed. The cracking occurred in the bolt head-to-shank fillet area and was attributed to IGSCC (a.k.a., PWSCC in nickel-base materials). The bolt stress levels were reportedly at the yield strength of the material.

Failures have been attributed to three factors:

1. Heat treatment condition
2. High peak stresses
3. Surface damage due to fabrication processes

Failures of Alloy X-750 clevis insert bolts were reported by one Westinghouse-designed plant in 2010. The lower clevis structure works with the radial keyways on the core barrel to provide rotational alignment for the lower internals. The Alloy X-750 bolting was used to fasten the Alloy 600 clevis inserts to the RV lugs. Although the failed clevis insert bolts were not removed for metallurgical examination, it can be surmised that the most likely cause of failure was

[1] Hot rolled, "equalized" at 1625°F (885°C) followed by 20 hours at 1300°F (704°C).

PWSCC. The clevis insert bolting had been heat treated in a condition similar to the AH treatment that has proven to be susceptible to PWSCC in the guide tube support pins. The relatively long time to failure in the clevis insert bolting may be attributed to the lower service temperature.

IASCC — A considerable amount of PWR internals IASCC has been observed in European PWRs since the 1980s, with emphasis on cracking of baffle-former bolting. Ultrasonic (UT) testing of baffle-former bolts in six French PWRs discovered failure rates ranging from 1.2% to 11% of the 960 total bolts. For this reason, the U.S. PWR owners and operators began a program to inspect the baffle-former bolting in order to determine whether similar problems might be expected in U.S. plants. One benefit of this program was the experience gained with the UT examination techniques used in the inspections. In addition, the industry began laboratory testing projects in order to gather the materials data necessary to support future inspections and evaluations.

As part of the U. S baffle-former bolt program, UT inspections were performed at two units with cold-worked Type 316 stainless steel bolting (1998/1999). In one unit, 1086 of 1088 bolts were inspected with no indications. Two bolts could not be inspected due to accessibility and were replaced. In the second unit, all 1088 bolts were inspected, again with no indications. A proactive minimum bolt pattern replacement was performed at these plants (276 bolts for Unit 1 and 203 bolts for Unit 2). Bolts removed from these plants were subject to follow-on mechanical testing and hot cell examination. This follow-on testing confirmed the NDE results.

The program also included inspection of two plants with solution-annealed Type 347 stainless steel baffle-former bolting. In one plant, all 728 baffle-former bolts were inspected by UT in 1998, with 55 bolts (7.5%) having indications that exceeded the UT acceptance criteria. At another unit, 639 out of the 728 solution-annealed Type 347 stainless steel baffle-former bolts were examined in 1999, with 59 bolts (9.2%) having indications failing to meet the UT acceptance criteria. At the first unit, on-site underwater mechanical testing of the removed baffle-former bolts indicated that the actual number of defective bolts was lower than suggested by the UT inspection. However, these known European or domestic baffle-former bolt IASCC indications are not necessarily applicable to all PWR designs. To date, the incidents have been generally associated with cold-worked Type 316 stainless steel or solution-annealed Type 347 stainless steel.

Bolts fabricated from solution-annealed Type 304 stainless steel appear to be less susceptible. An inspection was performed at one B&W-designed unit in 2005 on all 864 baffle-former bolts and UT indications were not observed.

In 2010, one Westinghouse plant reported finding several broken Type 347 stainless steel baffle-former bolt heads and Type 304 stainless steel locking bars on the lower core plate during a normal refueling outage. Subsequent investigation identified a region containing approximately 40 broken or severely damaged bolts. The damage was limited to the upper half of a single baffle plate.

Baffle-former bolt inspections have been conducted under the guidance of MRP-227 at three different Westinghouse-design U.S. domestic plants. The original bolting material at all three plants is Type 347 stainless steel; though, one of the units did replace a subset of the bolts in 1999 with Type 316 stainless steel.

The first of these inspections was a full UT examination of the baffle-former bolts conducted in 2010. The UT inspection detected one likely flaw out of 1088 bolts. Additionally, visual inspection of the baffle-former assembly detected two baffle-former bolts with missing lock bar welds. Each lock bar should have two welds to hold it in place, and these two bolts only had one weld. The missing welds were dispositioned as fabrication errors. The missing welds and flawed bolt were left in service.

The second inspection was a full UT examination of the baffle-former bolts conducted in 2011. The UT inspection detected two likely flaws out of 1088 bolts. Visual examination of the baffle-former assembly did not detect any reportable indications. The flawed bolts were left in service.

The one unit which replaced a subset of bolts in 1999 conducted a UT examination on a subset of bolts. All of the 1999 replacement bolts were inspected and approximately 100 of the original bolts were inspected. Additionally, a small number of bolts were removed and replaced. When possible, these bolts were inspected by UT after removal. Of the bolts inspected, only one defective bolt was detected. This bolt was left in service.

Flow-induced Vibration — In the earlier PWRs, a number of incidents occurred indicating that thermal shields and their support system could be vulnerable to the high flow forces in the vessel-core barrel downcomer. Westinghouse, CE, and B&W responded to these experiences in different ways. The Westinghouse approach was to add vibration-resistance to the shields and to embark on a program to develop advanced thermal shield designs for future plants. For CE plants, thermal shields were removed from operation for all but one facility, which has maintained integrity through positioning pin replacement, tightening, and inspection. The B&W approach was to modify and repair the thermal shields for improved resistance to vibration.

The dominant degradation mechanisms in thermal shields are high-cycle fatigue and SCC resulting from flow-induced vibration, with mechanical wear as a potential consequence. These degradation events appeared predominantly in the earliest thermal shield designs. Typically, the degraded components were fasteners or thermal shield support structures, not the thermal shield itself.

Two CE plants reported cases where failures in the thermal shield resulted in damage to the core barrel. The thermal shields were removed from both plants and the damage to the core barrels was mitigated.

Three early Westinghouse plants identified thermal shield degradation. The thermal shield degradation in these three plants was repaired; however, they are no longer operating and no operating plant has the same thermal shield design. Two additional Westinghouse plants have reported isolated failures of core barrel bolting that may be linked to flow-induced vibration.

Loss of Material

Wear — Wear of the in-core instrumentation thimble tubes was observed in the top part of the Zircaloy-4 thimble tubes at three CE-designed units. These tubes experienced through-wall tube degradation as a result of flow-induced vibration in the vicinity of the fuel alignment plate. This particular wear phenomenon was addressed by making modifications to the fuel alignment plate to alter the flow conditions in the vicinity of the entry point of the thimble tubes into the plate. Wear as a result of flow-induced vibration has not been observed in these components after

implementing the modifications to the fuel alignment plate. Accordingly, these components are not considered susceptible to this type of wear in the future.

Problems were noted involving the original locking devices for the B&W-design vent valve jackscrews in the late 1970s and early 1980s. The jackscrew locking mechanism was vibrating and wearing through the locking cup. A new locking mechanism was designed and supplied to most B&W units. At least four of the eight vent valves were modified with the redesigned locking devices. The four vent valves next to the two outlet nozzles were replaced. In the late 1970s and early 1980s, problems were also noted involving the original jackscrew guide bushing, which was found to be improperly secured on some valves. Procedures were developed to install the modified locking device on the jackscrew and to secure the bushing when necessary.

Wear of the Westinghouse control rod guide tube assembly guide cards has been reported at several domestic and international plants. The wear enlarges the guide card holes that guide the control rods through the assembly and maintain the alignment of the rods. A program is currently in progress through the PWROG to establish guidelines for managing this wear.

The wear surfaces on the radial keyways and clevis inserts are routinely examined as part of the PWR internals ASME B&PV Code Section XI inservice inspection programs. While reports of scratches, superficial wear, or both are common in these inspections, one European plant has reported significant wear scars at these surfaces. Efforts to establish quantitative acceptance criteria are ongoing.

In all currently operating Westinghouse and B&W plants, the incore flux detectors are directed through the RV bottom head via thimble tubes or guideways. For the bottom-mounted instrumentation design, the thimble tubes are retractable, and the insertion and retraction of these tubes are directed by long-radius guides below the bottom head and by internals guides between the bottom head and fuel assemblies. There is significant variation among plants with regard to thimble tube diameters (outer and inner), thimble tube-to guide path clearance, length of thimble tube exposed to coolant, and flow conditions.

The primary historical concerns with flux thimble degradation in Westinghouse-designed plants have been obstruction of the flux detector pathways, wear due to flow-induced vibration of the thimble tube, flow-induced vibration fatigue damage to thimble tube guideways, and damage to in-core instrumentation flange seating surfaces at refueling. The obstruction problem can often be mitigated by appropriate cleaning procedures at refueling. All Westinghouse plants are required by NRC Bulletin 88-09 to have an inspection program to periodically confirm incore neutron monitoring system thimble tube integrity. Reductions in wall thickness due to wear are normally monitored with an eddy-current inspection. Many plants have chosen to replace the flux thimbles with improved designs. These programs have been successful in managing thimble tube degradation.

A visual inspection in 1973 at one CE-designed plant revealed worn areas in the RV flange and head resulting from inadequate hold-down spring design and subsequent PWR internals vibration. Prior to shutdown, higher than normal ex-core neutron detector readings had suggested the possibility of excessive internals vibration. Wear was found on the mating surfaces, alignment keys and slots, snubbers, and outlet nozzle faces. The worn surfaces were repaired and a new design using Belleville spring assemblies greatly increased hold-down capacity and mitigated the issue.

In the Westinghouse-design and in two CE-designed units, PWR internals hold-down rings (or springs) were fabricated with Type 304 stainless steel. The subsequent CE-designed units switched to a modified Type 403 stainless steel hold-down ring, which shows less reduction in preload over the lifetime of the component. At least one international Westinghouse-designed plant has replaced their Type 304 stainless steel hold-down rings. Those that have not are managing potential degradation through physical measurement.

Change in Dimension

Irradiation-Induced Growth — Although irradiation-induced growth of zirconium alloys in CE-designed plants was not explicitly identified in MRP-175 as an age-related degradation mechanism to be evaluated as part of the screening process, irradiation-induced growth in the axial direction of the in-core instrumentation thimble tubes has reduced the clearance between the thimble nose and the bottom of the fuel assembly. Some plants had observed that the thimble tube support plate was raised above its normal support position when the upper internals structure was set in place after fuel reload. This indicated that for some of the thimbles the gap tolerance between the thimble tube and the bottom end fitting of the fuel assembly had been reduced until the tube contacted the bottom fitting of the fuel assembly and was being loaded in compression. Ten plants affected by this issue have taken actions. Six of these plants have already replaced the thimble tube assemblies with modified designs that are shorter in length to accommodate the expected irradiation-induced growth. Two additional plants have replacement designs in fabrication and have made preparations to install the replacement thimbles in an upcoming outage. The remaining two plants have not yet begun preparations for a full replacement of the thimble tubes, but one of these two has instead taken the intermediate step of raising the thimble support plate to accommodate additional axial growth. These plants are planning to execute a thimble assembly replacement program during a future refueling outage that is not currently encumbered with other large-scale replacements of major components. All affected plants will likely have replaced their thimble tubes prior to license extension.

Miscellaneous

B&W-design Vent Valves — Vent valve jackscrew locking cup damage has also been observed at some units, which was due to an interaction with the plenum assembly during insertion and removal activities. Vent valves are replaceable items and as noted above, have been replaced as necessary.

Mechanism Unidentified to Date — Visual examinations at one B&W-designed unit in 2005 indicated that three or four internal baffle-to-baffle bolts were found protruding. The bolt heads extended beyond the baffle plate surface. This was an indication that the locking devices, and potentially the bolts as well, had failed. As noted above, a UT inspection of 100% of the baffle-former bolts was performed, with no detected indications of broken bolts. No UT inspection was performed on the internal baffle-to-baffle bolts, and the potentially failed baffle-to-baffle bolts have yet to be removed to confirm failure and, if failed, the mechanism. As a result of the observations, AREVA performed a unit-specific evaluation to assess operational and safety functions for continued operation. That evaluation included thermal hydraulic evaluation, structural evaluation, fuel evaluation, and loose parts evaluation.

RVI Component Replacements — Replacement of upper internals in Westinghouse and CE designs have been made.

Beginning in 2004, replacement of the complete internals (upper and lower internals) at three Japanese PWRs has also been performed. It has been stated that these replacements have been performed for the following reasons:

1. To keep and improve operational reliability, safety, and a high load factor for the nuclear power units
2. To maintain the plant against aging degradation of the PWR internals
3. To mitigate degradation risks that would rise with increasing operational time in the future

B REQUESTS FOR ADDITIONAL INFORMATION AND MRP RESPONSES

May 28, 2009

Mr. Christian B. Larsen
Nuclear Vice President & Chief Officer
Electric Power Research Institute
3420 Hillview Avenue
Palo Alto, CA 94304-1338

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION RE: ELECTRIC POWER RESEARCH INSTITUTE TOPICAL REPORT 1006596, "MATERIALS RELIABILITY PROGRAM: PRESSURIZED WATER REACTOR INTERNALS INSPECTION AND EVALUATION GUIDELINES (MRP-227-REV. 0)" (TAC NO. ME0680)

Dear Mr. Larsen:

By letter dated January 12, 2009, Electric Power Research Institute (EPRI) submitted for U.S. Nuclear Regulatory Commission (NRC) staff review Topical Report (TR) 1006596, "Materials Reliability Program (MRP): Pressurized Water Reactor Internals Inspection and Evaluation Guidelines MRP-227-Rev. 0)." Upon review of the information provided, the NRC staff has determined that additional information (RAI) is needed to complete the review. During a conference call on May 21, 2009, the NRC staff requested that additional EPRI reports referenced in TR MRP-227 be provided expeditiously to the NRC staff. The NRC staff is providing its request for these additional EPRI reports in the enclosed RAI. Please note that a second set of RAIs will be issued separately to capture the technical questions related to the NRC staff's review of TR MRP-227. During a conference call on May 21, 2009, Ann Deema, Senior Project Manager, and I agreed that the NRC staff will receive your response to the enclosed RAI questions by June 12, 2009.

If you have any questions regarding the enclosed RAI questions, please contact me at 301-415-3610.

Sincerely,

/RA/

Tanya M. Mensah, Senior Project Manager
Special Projects Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project Nos. 669 and 689

Enclosure: Request for Additional Information

cc w/encl: See next page

Mr. Christian B. Larsen
Nuclear Vice President & Chief Officer
Electric Power Research Institute
3420 Hillview Avenue
Palo Alto, CA 94304-1338

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If you have any questions regarding the enclosed RAI questions, please contact me at 301-415-3610.

Sincerely,

/RA/

Tanya M. Mensah, Senior Project Manager
Special Projects Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project Nos. 669 and 689

Enclosure: Request for Additional Information

cc w/encl: See next page

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ADAMS ACCESSION NO.: ML091460086 **NRR-106**

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Project No. 689
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REQUEST FOR ADDITIONAL INFORMATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT 1006596, "MATERIALS RELIABILITY PROGRAM: PRESSURIZED WATER REACTOR INTERNALS INSPECTION AND EVALUATION GUIDELINES (MRP-227-REV. 0)

ELECTRIC POWER RESEARCH INSTITUTE

PROJECT NO. 669

During a conference call on May 21, 2009, the NRC staff requested that the following Electric Power Research Institute (EPRI) documents be submitted expeditiously to the NRC to support the staff's review of Topical Report (TR) Materials Reliability Program (MRP)-227.

(1) MRP-189, Revision 1, "Materials Reliability Program: Screening, Categorization, and Ranking of B& W-Designed PWR Internals."

(2) MRP-190, "Materials Reliability Program: Failure Modes, Effects, and Criticality Analysis of B&W-Designed PWR Internals."

(3) MRP-191, "Materials Reliability Program: Screening, Categorization and Ranking of Reactor Internals of Westinghouse and Combustion Engineering PWR Designs."

(4) MRP-210, "Materials Reliability Program: Fracture Toughness Evaluation of Highly Irradiated PWR Stainless Steel Internal Components."

(5) MRP-229, "Materials Reliability Program: Functionality Analysis for B&W Representative PWR Internals."

(6) MRP-230, "Materials Reliability Program: Functionality Analysis for Westinghouse and Combustion Engineering Representative PWR Internals."

(7) MRP-231, "Materials Reliability Program: Aging Management Strategies for B&W PWR Internals."

(8) MRP-232, "Materials Reliability Program: Aging Management Strategies for Westinghouse and Combustion Engineering PWR Internals."

Based upon this conference call, the NRC staff understood that the EPRI would provide both proprietary and non-proprietary versions of these documents, with the exception of MRP-229 and MRP-230. The NRC staff and EPRI agreed during the conference call that these two documents were currently not needed by the NRC staff. However, if the NRC staff determines at a later date that they need to review MRP-229 and MRP-230 to support the review of MRP-227, then the EPRI will submit proprietary and non-proprietary versions of these documents expeditiously to the NRC Document Control Desk.

ENCLOSURE

ELECTRIC POWER
RESEARCH INSTITUTE

MRP Materials Reliability Program_____MRP 2009-043

(via email)

June 10, 2009

Tanya M. Mensah
U.S. Nuclear Regulatory Commission
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SUBJECT: REQUEST FOR ADDITIONAL INFORMATION RE: ELECTRIC POWER RESEARCH INSTITUTE TOPICAL REPORT 1006596,'MATERIALS RELIABILITY PROGRAM: PRESSURIZED WATER REACTOR INTERNALS INSPECTION AND EVALUATION GUIDELINES (MRP-227-REV. 0)' (TAC NO. ME0680)

Reference:

1. Letter Tanya Mensah (NRC) to Christian B. Larsen (EPRI), same subject, dated May 28, 2009
2. Letter and Affidavit, Christian B. Larsen (EPRI) to NRC Document Control Desk, Request for Withholding Commercial Documents, dated June 10, 2009

Dear Ms. Mensah:

In response to your May 28 letter (Reference 1) requesting that EPRI provide copies of reports to support NRC review of EPRI Report 1006596, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-Rev. 0)" we are forwarding eight copies of the following four documents:

1) *Materials Reliability Program:* Screening, Categorization, and Ranking of B&W-Designed PWR Internals Component Items (MRP-189-Rev. 1). EPRI, Palo Alto, CA: 2009. 1018292;

2) *Materials Reliability Program:* Fracture Toughness Evaluation of Highly Irradiated PWR Stainless Steel Internal Components (MRP-210). EPRI, Palo Alto, CA: 2007. 1016106;

3) *Materials Reliability Program:* Aging Management Strategies for B&W PWR Internals (MRP-231). EPRI, Palo Alto, CA: 2008. 1016592;

4) *Materials Reliability Program:* Aging Management Strategies for Westinghouse and Combustion Engineering PWR Internals (MRP-232). EPRI, Palo Alto, CA: 2008. 1016593.

These documents have been forwarded to the Document Control Desk by Reference 2 (copy attached) requesting that this copyrighted information be withheld from public disclosure. Two of the requested documents, MRP-190, "Materials Reliability Program: Failure Modes, Effects, and Criticality Analysis of B&W-Designed PWR Internals" and MRP-191, "Materials Reliability Program: Screening, Categorization and Ranking of Reactor Internals of Westinghouse and Combustions Engineering PWR Designs" have been made publicly available and are no longer controlled documents. They may be downloaded, at no cost, from the EPRI web site (www.EPRI.com).

As discussed during the conference call on May 21, 2009, AREVA has identified a minor error in the functionality analysis for the B&W plants that is reflected in one of the documents forwarded with this transmittal, MRP-231. The error does not affect the aging management strategies contained in MRP-231 nor the related recommendations in MRP-227; however, for completeness and accuracy the error is being corrected and will result in a revision to MRP-231. At such time as MRP-231, Rev. 1 is available, it will be provided to the NRC under a separate cover letter.

If you have any questions, please contact Christine King at 650-855-2605, or Anne Demma at 650-855-2026.

Best Regards,

Dennis P. Weakland

Dennis Weakland
FirstEnergy Nuclear Operating Co.
Chairman, Materials Reliability Program

Cc: Mike Melton (NEI)
Don Dyksterhouse (INPO)
Chris Larsen (EPRI)
Christine King (EPRI)
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NEI Project Nos. 669 and 689

August 24, 2009

Mr. Christian B. Larsen
Nuclear Vice President & Chief Officer
Electric Power Research Institute
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SUBJECT: REQUEST FOR ADDITIONAL INFORMATION RE: ELECTRIC POWER RESEARCH INSTITUTE TOPICAL REPORT 1006596, "MATERIALS RELIABILITY PROGRAM (MRP): PRESSURIZED WATER REACTOR INTERNALS INSPECTION AND EVALUATION GUIDELINES (MRP-227 – REV. 0) (TAC NO. ME0680)

Dear Mr. Larsen:

By letter dated January 12, 2009, Electric Power Research Institute submitted for U.S. Nuclear Regulatory Commission (NRC) staff review Topical Report (TR) 1006596, "Materials Reliability Program (MRP): Pressurized Water Reactor Internals Inspection and Evaluation Guidelines." Upon review of the information provided, the NRC staff has determined that additional information (RAI) is needed to capture the initial set of technical questions related to the NRC staff's review of TR MRP-227 to support completion of the review. The NRC staff will however issue another set of RAIs based on its review of the MRP-227 report and its supporting reports. By e-mail dated August 22, 2009, Ms. Christine King, Program Manager, MRP, and I agreed that the NRC staff will receive your response to the enclosed RAI questions within 60 days of issuance of this letter. If you have any questions regarding the enclosed RAI questions, please contact me at 301-415-3610.

Sincerely,

/RA/

Tanya M. Mensah, Senior Project Manager
Special Projects Branch
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Office of Nuclear Reactor Regulation

Project Nos. 669 and 689

Enclosure:
RAI questions

cc w/encl: See next page

August 24, 2009

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SUBJECT: REQUEST FOR ADDITIONAL INFORMATION RE: ELECTRIC POWER RESEARCH INSTITUTE TOPICAL REPORT 1006596, "MATERIALS RELIABILITY PROGRAM (MRP): PRESSURIZED WATER REACTOR INTERNALS INSPECTION AND EVALUATION GUIDELINES (MRP-227 – REV. 0) (TAC NO. ME0680)

Dear Mr. Larsen:

By letter dated January 12, 2009, Electric Power Research Institute submitted for U.S. Nuclear Regulatory Commission (NRC) staff review Topical Report (TR) 1006596, "Materials Reliability Program (MRP): Pressurized Water Reactor Internals Inspection and Evaluation Guidelines." Upon review of the information provided, the NRC staff has determined that additional information (RAI) is needed to capture the initial set of technical questions related to the NRC staff's review of TR MRP-227 to support completion of the review. The NRC staff will however issue another set of RAIs based on its review of the MRP-227 report and its supporting reports. By e-mail dated August 22, 2009, Ms. Christine King, Program Manager, MRP, and I agreed that the NRC staff will receive your response to the enclosed RAI questions within 60 days of issuance of this letter. If you have any questions regarding the enclosed RAI questions, please contact me at 301-415-3610.

Sincerely,
/RA/

Tanya M. Mensah, Senior Project Manager
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Project Nos. 669 and 689

Enclosure:
RAI questions

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REQUEST FOR ADDITIONAL INFORMATION (RAI)

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT 1006596, "MATERIALS RELIABILITY PROGRAM (MRP): PRESSURIZED WATER REACTOR INTERNALS INSPECTION AND EVALUATION GUIDELINES

(MRP-227 – REV. 0)

ELECTRIC POWER RESEARCH INSTITUTE

PROJECT NO. 669

In a letter dated January 12, 2009, the Electric Power Research Institute (EPRI) submitted a Topical Report (TR) MRP-227, "Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines," which addresses the development of an aging management program (AMP) for PWR reactor vessel internal (RVI) components. On July 2, 2009, EPRI provided additional reports that support the technical bases used for developing the AMP, and these reports were submitted to the NRC staff for information only. The NRC staff has reviewed TR MRP-227and developed an initial set of RAIs. Based on further review of the supporting reports, the NRC staff may issue additional RAIs at a later date.

RAI-1 Many components are placed on a standard 10-year inservice inspection interval coincident with typical American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) inspection requirements. It's not clear, however, whether this 10-year interval is technically acceptable for PWR RVI components. No justification in light of the specific degradation mechanisms being managed has been provided. Other inspection intervals and requirements are based on a certain number of operating cycles. The acceptability of these intervals has also not been established. Please provide a technical justification of the intervals chosen relative to the mechanisms being managed in TR MRP-227.

RAI-2 In Tables 4-1 through 4-6 and Tables 4-8 and 4-9 of TR MRP-227, the MRP intends to implement visual testing (VT-3) examinations to identify cracking in some PWR RVI components. Historically, enhanced visual testing (EVT-1) or ultrasonic testing (UT) methods are used to effectively identify cracks. Explain why the use of a VT-3 inspection method should be considered acceptable for identifying cracking in some PWR RVI components.

RAI-3 Eddy current testing (ET) is identified in TR MRP-227 as an inspection method to be used to identify cracking in some PWR RVI components. Clarify whether the acceptance criterion for ET inspections will be based on a "pass – no pass" acceptance criterion (i.e., any ET signals indicating a relevant ET indication would fail the acceptance criterion).

RAI-4 The accessibility of the primary inspection RVI components is not typically addressed. It is therefore not clear how much inspection coverage is necessary to ensure timely detection of aging effects in the primary inspection RVI components. Discuss whether guidance should be provided in TR MRP-227 regarding minimum inspection volumes/areas which must be achieved to take credit for having effectively inspected a particular RVI component.

ENCLOSURE

RAI-5 During the extended period of operation, some PWR RVI components are subject to high levels of neutron radiation which may lead to irradiation embrittlement and a loss of fracture toughness and the potential for irradiation-assisted stress corrosion cracking. In combination, these effects may lead to the potential for component failure under some design basis loading conditions. Explain how licensees will be expected to account for potential reduction in fracture toughness when evaluating cracks that are detected during the required inspections, in particular when establishing the frequency of subsequent inspections after cracking is identified.

RAI-6 Loose parts could be generated due to deterioration of some PWR RVI components during the extended period of operation. Provide information which addresses how the following consequences of loose parts generation were considered in development of the inspection program given in TR MRP-227.

(a) potential for fuel bundle flow blockage and consequential fuel damage,

(b) potential for interference with control rod operation, and

(c) potential for impact damage on reactor internals.

RAI-7 Alloy 600 PWR RVI components and their associated welds manufactured from Alloys 82 and 182 are susceptible to primary water stress corrosion cracking (PWSCC) when exposed to PWR reactor coolant water. In Table 3-1 of TR MRP-227, the following Babcock and Wilcox (B&W) Alloy X-750 PWR RVI components were welded with Alloy 82 material and yet they were classified under "N" category which excludes inspections for these PWR RVI components:

(1) dowel-to-core barrel cylinder welds, (2) dowel-to-upper grid rib section bottom flange welds, (3) dowel locking welds, (4) dowel-to-guide block welds, and (5) dowel-to-distributor flange welds.

Even though stress levels in these components may not exceed the threshold levels, the NRC staff considers it to be likely that PWSCC can potentially occur due to the introduction of cold work during fabrication. In light of this observation, provide an explanation for excluding inspection requirements for these B&W PWR RVI components.

RAI-8 When exposed to a light-water reactor temperatures of approximately 500 °F or higher, the 17-4 precipitation hardened (PH) martensitic stainless steel (MSS) that has previously been subjected to aging (heat treatment) at about 1100 °F can experience thermal embrittlement and an increase in hardness and a reduction in Charpy V-notch impact test toughness. Operating experience from Oconee Nuclear Station (Information Notice (IN) 2007-02, ADAMS Accession Number ML070100459) shows that thermally embrittled 17-4 PH MSS is susceptible to failure when exposed to unexpected loading conditions. In IN 2007-02, the NRC staff recommended that licensees prevent the deleterious effects of thermal embrittlement in the 17-4 PH MSS components by identifying aging degradation (i.e., cracks), implementing early corrective actions, and monitoring and trending age-related degradation. Therefore, the NRC staff requests that the TR MRP-227 report should include thermal embrittlement as an aging effect for any 17-4 PH MSS RVI components.

RAI-9 With respect to the management of cast austenitic stainless steel (CASS) aging and embrittlement TR MRP-227 does not appear to address the program's compliance with the requirements specified in the relevant Generic Aging Lessons Learned (GALL) Report AMPs. Provide a discussion of how TR MRP-227 adequately addresses the requirements specified in GALL AMP, XI.M12, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)," and GALL AMP XI.M13, "Thermal Aging and Neutron Embrittlement of Cast Austenitic Stainless Steel (CASS)," for CASS materials used in PWR RVI components. Alternatively, if the management CASS PWR RVI component aging is not treated within the scope of TR MRP-227, provide a proposed modification of the report which documents how licensees are expected to manage this mechanism outside of the TR MRP-227 program.

RAI-10 According to Section A.1.4 in MRP-175, "Materials Reliability Program: PWR Internal Aging Degradation Mechanism Screening Threshold Values," susceptibility to SCC in nickel-based Alloy X-750 PWR RVI components depends on the type of heat treatment that is performed on the alloy. High temperature heat treatment processes that are used on Alloy X-750 components offer better resistance to SCC than the other age hardened heat treatment processes. Licensee determination of the heat treatment applied to their Alloy X-750 PWR RVI components would appear to be a critical parameter in ensuring the licensee's AMP will adequately manage the potential effects of aging. Discuss whether this determination should be included as a license renewal application action item.

RAI-11 Following on to RAI-10, additional aspects of the TR MRP-227 methodology may need to be addressed by license renewal applicant action items for applications currently under review or those that have yet to be submitted to the NRC. The NRC staff requests the MRP's assistance in identifying potential action items which are: (1) necessary to provide plant-specific information to complete the AMP; (2) necessary to confirm applicant compliance with important assumptions underlying the MRP-227 methodology; or (3) other considerations.

RAI-12 Provide the loading combinations that were used in determining the peak stress values for any given PWR RVI component. The NRC staff believes that plants that have been implementing power uprates will have to assess whether the peak stress values for any given PWR RVI component are affected by power uprate conditions to determine if their plant is bounded by the assumptions underlying TR MRP-227.

RAI-13 Certain degradation mechanisms (e.g., void swelling in B&W PWR RVI components) are not inspected for in a particular reactor type. Why does the program not require the most susceptible location for each mechanism in each reactor-type (i.e., B&W, Combustion Engineering, or Westinghouse) be inspected as a primary component to insure that each degradation mechanism is not occurring within the reactor?

RAI-14 Discuss how the PWR RVI components in each reactor design considered to be the most susceptible to (or most likely to first demonstrate the effects of) a particular degradation mechanism did, or did not, get binned in the primary inspection component group for that design.

RAI-15 The failure modes, effects, and criticality analysis (FMECA) uses a probabilistic approach with regard to structural stability of any given RVI component and includes the development of a "failure probability factor". What methodology was used to establish the failure probability factor of any RVI component?

RAI-16 Clarify the conditions under which design basis event (DBE) effects on component performance were considered. How does this approach provide reasonable assurance that the margins against failure are adequately maintained during the license renewal period?

RAI-17 Component failure due to the same degradation mechanism is not considered to be a common cause failure because of the expectation that damage initiation and growth occurs at different times. However, certain DBEs could potentially lead to a plant condition (damage state) that would not occur unless multiple components were degraded. Discuss how the potential for multi-component failure due to a DBE was considered as part of the development of the MRP-227 program.

RAI-18 Clarify how plant-specific differences were considered within the FMECA. Discuss whether any additional plant-specific analyses are required, either as a supplement to TR MRP-227 or as indentified plant-specific action items, in order to assure that FMECA analysis supporting the TR MRP-227 program is applicable to a given facility.

RAI-19 Discuss how a licensee will demonstrate adherence to the reference core loading pattern on a unit-specific basis. Address plant-to-plant variability in neutron flux at various peripheral core locations. Confirm, based on significant operating experience, that "low-leakage" core designs, when normalized by power density, have peripheral neutron fluxes that are consistently within the estimates for the generically studied plants.

RAI-20 Provide a technical basis to justify the examination acceptance criteria, the sufficiency and relevancy of the links between primary and expansion group components (why were those particular links chosen), and the expansion criteria. Discuss also the technical basis that applied to place certain components in the primary category while others were placed in the expansion category.

RAI-21 Many of the acceptance criteria provided in TR MRP-227 are vague such as finding "detectable crack-like surface indications," or "damaged or fractured material," or "readily detectable cracking." It's not clear that these criteria will be uniformly interpreted or implemented from plant to plant. Discuss the need to develop more detailed acceptance criteria on a plant-specific basis and how will the sufficiency of these criteria be established.

RAI-22 The screening criteria groups materials into susceptibility levels for each degradation mechanism: highly susceptible, moderately susceptible, susceptible, and "below the screening criteria." Discuss the criteria used to distinguish among the different levels of susceptibility.

RAI-23 Discuss whether an evaluation was performed for any specific high consequence of failure PWR RVI components such that their inspection might be warranted even in the absence of a currently identifiable mechanism. Are there any PWR RVI components that should be monitored through in-service inspection to protect against unforeseen failure due to the emergence of a potential future degradation mechanism?

RAI-24 Relevant US and international operating experience with respect to RVI components is not summarized. It is important to indicate what prior RVI component inspections have identified, in particular with respect to justifying the adequacy of existing programs and as part of

the basis for the examination requirements (e.g., type, periodicity, importance) identified in MRP-227.

RAI-25 The cumulative usage factor values for several B&W components need to be confirmed during a comprehensive search of all existing stress and fatigue calculations for the PWR internals. Discuss how such items are intended to translate into plant-specific action items.

RAI-26 The implications of void swelling are indicated as "dimensional change and distortion…" and it is also noted that "severe void swelling may result in cracking under stress." However, it is not indicated that void swelling can lead to reduced fracture toughness in materials even though it is noted in Section 3.2.7 of TR MRP-227 that "severe swelling (>5%) has been correlated with extremely low fracture toughness values." It is not clear how much void swelling is needed before distortion is detectable via VT-3 examination in susceptible PWR RVI components and whether this threshold for dectectability will also address the concern over potential loss of fracture toughness due to void swelling. Provide a discussion of this topic.

February 1, 2010

Ms. Tanya M. Mensah
Senior Project Manager
Special Projects Branch, Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Subject: EPRI MRP Responses to: REQUEST FOR ADDITIONAL INFORMATION RE: ELECTRIC POWER RESEARCH INSTITUTE TOPICAL REPORT 1016596,'MATERIALS RELIABILITY PROGRAM: PRESSURIZED WATER REACTOR INTERNALS INSPECTION AND EVALUATION GUIDELINES (MRP-227-REV. 0)' (TAC NO. ME0680), August 24, 2009 (please note the corrected EPRI Product Number – 1016596 – from that in the actual RAI request letter – 1006596)

Reference:

1. Letter Tanya Mensah (NRC) to Christian B. Larsen (EPRI), Request for Additional Information, dated August 24, 2009

Dear Ms. Mensah:

In response to your August 24 letter (Reference 1) we are forwarding ten copies of the subject response, eight copies for the staff and two for the Document Control Desk.

If you have any questions, please contact Christine King at 650-855-2605, or Anne Demma at 650-855-2026.

Best Regards,

Terry McAlister
South Carolina Electric & Gas Co.
Chairman, Materials Reliability Program

cc: NRC Document Control Desk (with Attachment – 2 copies)
Victoria Anderson, NEI
William Greeson, INPO
Christine King, EPRI
Anne Demma, EPRI

Final Responses to the 2nd set of RAIs on MRP-227-Rev. 0
02/01/2010

Titles of MRP Reports Referenced in MRP-227-Rev. 0 or Referred to in RAI Responses

MRP #	Title	EPRI #
MRP-128	*Materials Reliability Program: Characterization of Decommissioned PWR Vessel Internals Material Samples – Material Certification, Fluence, and Temperature,* 2003	1008202
MRP-134	*Materials Reliability Program: Framework and Strategies for Managing Aging Effects in Reactor Internals,* 2005	1008203
MRP-135 - Rev. 1	*Materials Reliability Program: Development of Material Constitutive Model for Irradiated Austenitic Stainless Steel,* 2009	1018291
MRP-156	*Materials Reliability Program: Pressurized Water Reactor Issue Management Table, PWR-IMT, Consequence of Failure,* 2005	1012110
MRP-157	*Materials Reliability Program: Updated B&W Design Information for the Issue Management Tables,* 2005	1012132
MRP-175	*Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values,* 2005	1012081
MRP-189 - Rev. 1	*Materials Reliability Program: Screening, Categorization, and Ranking of B&W-Designed PWR Internals,* 2009	1018292
MRP-190	*Materials Reliability Program: Failure Modes, Effects, and Criticality Analysis of B&W-Designed PWR Internals,* 2006	1013233
MRP-191	*Materials Reliability Program: Screening, Categorization and Ranking of Reactor Internals of Westinghouse and Combustion Engineering PWR Designs,* 2006	1013234
MRP-210	*Materials Reliability Program: Fracture Toughness Evaluation of Highly Irradiated PWR Stainless Steel Internal Components,* 2007	1016106
MRP-211	*Materials Reliability Program: PWR Internals Age-Related Material Properties, Degradation Mechanisms, Models, and Basis Data – State of Knowledge,* 2007	1015013
MRP-228	*Materials Reliability Program: Inspection Standard for Reactor Internals,* 2009	1016609

MRP-229 - Rev. 1	*Materials Reliability Program: Functionality Analysis for B&W-Designed Representative PWR Internals,* 2009	1019090
MRP-230 - Rev. 1	*Materials Reliability Program: Functionality Analysis for Westinghouse & CE-Designed Representative PWR Internals,* 2009	1019091
MRP-231	*Materials Reliability Program: Aging Management Strategies for B&W-Designed PWR Internals,* 2008	1016592
MRP-232	*Materials Reliability Program: Aging Management Strategies for Westinghouse and Combustion Engineering PWR Internals,* 2008	1016593

RAI-1 Many components are placed on a standard 10-year inservice inspection interval coincident with typical American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) inspection requirements. It's not clear, however, whether this 10-year interval is technically acceptable for PWR RVI components. No justification in light of the specific degradation mechanisms being managed has been provided. Other inspection intervals and requirements are based on a certain number of operating cycles. The acceptability of these intervals has also not been established. Please provide a technical justification of the intervals chosen relative to the mechanisms being managed in TR MRP-227.

Response:

The supporting documentation for MRP-227, such as the aging management strategy reports MRP-231 and MRP-232, evaluated each Primary, Expansion, or Existing Programs component and the associated degradation effects to determine the timing of initial and subsequent examinations or other inspections. The evaluation considered available operating experience for those components and laboratory data for the component materials, as well as information derived from the functionality analyses documented in MRP-229 and MRP-230. Based upon these evaluations, unless information gained from the first round of augmented inspections demonstrates otherwise, the schedule for initial examinations and the 10-year periodicity of subsequent examinations provide confidence that the aging effects for those components will be managed adequately and the functionality of the components will continue to be maintained.

However, the intention is that MRP-227, through its proactive approach to managing aging effects in PWR internals components, is and will continue to be a living document. The results of the first and subsequent rounds of augmented examinations will be reported to the MRP (see Section 7 of MRP-227) and may lead to changes in the periodicity of subsequent examinations. For example, if more aggressive degradation effects are identified through these augmented examinations, or through other operating experience, the findings would be evaluated to determine what, if any, changes are needed in the MRP-227 requirements.

Two sets of investigations performed during the development of MRP-227 provide strong support for the adequacy of the chosen inspection intervals. First, as documented in MRP-229 and MRP-230, the time history analyses of the individual and combined degradation mechanisms modeled in the functionality analyses showed a gradual, rather than a sudden, progression of potential degradation, with the majority of those aging effects accumulating during the conservatively assumed 30-years of high leakage core loading. The functionality analyses account for the combined effects of aging degradation mechanisms by including the irradiation/temperature-induced effects on material response, e.g. mechanical property changes, void swelling, creep/relaxation, etc., in the time history analysis. Second, the operating history review discussed in MRP-231 and MRP-232 confirms the finding that the required 10-year ASME Code Examination Category B-N-3 inspections and other, voluntary U.S. and international industry baffle-to-former bolt examinations show that 10-year examination intervals are appropriately conservative.

RAI-2 In Tables 4-1 through 4-6 and Tables 4-8 and 4-9 of TR MRP-227, the MRP intends to implement visual testing (VT-3) examinations to identify cracking in some PWR RVI components. Historically, enhanced visual testing (EVT-1) or ultrasonic testing (UT) methods are used to effectively identify cracks. Explain why the use of a VT-3 inspection method should be considered acceptable for identifying cracking in some PWR RVI components.

Response: The VT-3 examination was specified to identify general degradation in the aged components. In none of the cases where VT-3 is specified is the examination objective the detection of the onset of cracking with accompanying tight crack opening displacements (CODs). In the cases where cracking associated with IASCC/SCC is the anticipated mechanism or one of the mechanisms postulated and VT-3 is specified, the objective of the examination is detection of:

- Broken or missing bolt locking devices and welds
- Protruding bolts
- Broken or missing pieces (e.g., supports, spider arms, dowels)

In these cases, VT-3 as currently defined in Section XI of the ASME Code (0.106" character height resolution) is quite capable of this level of detection. This practice was deemed appropriate for the existing examination requirements listed in MRP-227 Tables 4-8 and 4-9.

As used in MRP-227, VT-3 is consistent with Section XI rules for pipe supports looking for:

- Missing bolts
- Gross degradation
- Misalignment
- General structural condition

Table RAI-2-1 provides a summary of the Primary and Expansion components from the B&W design where VT-3 is specified for cracking. In each case, the general condition of concern has been identified. The VT-3 inspections have been for cases where extensive cracking can occur without threatening the structural integrity of the internals. The use of VT-3 examinations to monitor cracking in the B&W internals can be summarized in four categories:

- Bolt locking devices (stress corrosion cracking, SCC), specified as separated or missing locking devices or welds
- Core Support Shield components (irradiation or thermal embrittlement, IE/TE), specified as detection of surface irregularities such as damaged, fractured, or missing material
- Baffle plates (irradiation embrittlement, IE), specified as "readily detectable" cracking
- IMI spiders/spider arms (TE/IE), specified as fractured or missing spider arms or separation of spider arms from the lower grid rib section at the weld

Table RAI-2-2 provides a summary of the of the Primary and Expansion components from the CE design where VT-3 is specified for cracking. There are only two CE components in this category:

- Instrument guide tubes attached to the upper CEA structure and the welds in the lower support structure. The inspection for this component specifies the concern as cracking that results in missing supports or separation at the welded joint between the tubes and the supports
- Core support column welds. The inspection for this component specifies the concern as damaged or fractured material.

Table RAI-2-3 provides a summary of the of the Primary and Expansion components from the Westinghouse design where VT-3 is specified for cracking. There are only three Westinghouse components in this category:

- Baffle-Edge Bolts. Although these bolts have a purpose, they do not carry primary loads required for structural integrity. The inspection is required to detect the following conditions:
    - Lost or broken locking devices
    - Failed or missing bolts
    - Protrusion of bolt heads.
- Thermal Shield Flexures. The inspection is required to detect excessive wear, fracture, or complete separation.
- Bottom Mounted Instrumentation (BMI) columns. The inspection is required to detect completely fractured column bodies

VT-3 examinations for cracking were not specified in cases where the data would potentially be used in a fracture mechanics analysis to demonstrate the structural integrity of the vessel internals. These more sensitive crack detection capabilities of the UT and EVT-1 examinations have been specified for other components where it was determined that early detection and protection against fracture was critical. However even in these cases the example calculations performed in MRP-210, demonstrate that the internals structures are extremely flaw-tolerant. See for example, the critical flaw sizes for the postulated stresses for a through-wall edge crack in a flat plate (Figure 3-18 and Table 3-3 in MRP-210). Thus, a VT-3 examination is capable of identifying subcritical crack growth well before a crack would become critical. For justification, see for example the final paragraph of Section 3.2.3.1 of MRP-231.

The MRP-227 inspection recommendations for VT-3 to monitor for the effects of cracking are appropriate because they are limited to cases where the intent of the examination is to monitor the general condition of the component. These recommendations are consistent with the approach used in the ASME Section XI examinations, which require VT-3 inspections for accessible core support structures. The practice is also consistent with the approach used in BWR applications. Further discussion of the inspection strategies for these components may be found in MRP-231 and MRP-232.

Table RAI-2-1 Primary and Expansion Components from B&W Designed Plants with Cracking Identified as an Effect and VT-3 Examinations Specified

Item	Effect (Mechanism)	Examination Method/Frequency	Examination Coverage
Primary **Core Support Shield Assembly** CSS cast outlet nozzles	Cracking (TE), including the detection of surface irregularities, such as damaged or fractured material	Visual (VT-3) examination during the next 10-year ISI. Subsequent examinations on the 10-year ISI interval.	100% of accessible surfaces. See Figure 4-9
Primary **Core Support Shield Assembly** CSS vent valve discs (Note 1)			100% of accessible surfaces. (See BAW-2248A, page 4.3 and Table 4-1.) See Figures 4-10 and 4-11
Primary **Core Support Shield Assembly** CSS vent valve top retaining ring CSS vent valve bottom retaining ring CSS vent valve disc shaft or hinge pin (Note 1)	Cracking (TE), including the detection of surface irregularities, such as damaged, fractured material, or missing items	Visual (VT-3) examination during the next 10-year ISI. Subsequent examinations on the 10-year ISI interval.	100% of accessible surfaces (See BAW-2248A, page 4.3 and Table 4-1.) See Figures 4-10 and 4-11
Primary **Core Support Shield Assembly** Upper core barrel (UCB) bolts and their locking devices	Cracking (SCC)	Volumetric examination (UT) of the bolts within two refueling outages from 1/1/2006 or next 10-year ISI interval, whichever is first. Subsequent examination to be determined after evaluating the baseline results. Visual (VT-3) examination of bolt locking devices on the 10-year ISI interval.	100% of accessible bolts. See Figure 4-7
Primary **Core Barrel Assembly** Lower core barrel (LCB) bolts and their locking devices	Cracking (SCC)	Volumetric examination (UT) of the bolts during the next 10-year ISI interval from 1/1/2006. Subsequent examination to be determined after evaluating the baseline results. Visual (VT-3) examination of bolt locking devices on the 10-year ISI interval.	100% of accessible bolts See Figure 4-8
Primary **Core Barrel Assembly** Baffle plates	Cracking (IE), including the detection of readily detectable cracking in the baffle plates	Visual (VT-3) examination during the next 10-year ISI. Subsequent examinations on the 10-year ISI interval.	100% of the accessible surface within 1 inch around each flow and bolt hole See Figure 4-2

Item	Effect (Mechanism)	Examination Method/Frequency	Examination Coverage
Primary **Core Barrel Assembly** Locking devices, including locking welds, of baffle-to-former bolts and internal baffle-to-baffle bolts	Cracking (IASCC, IE, Overload), including the detection of missing, non-functional, or removed locking devices or welds	Visual (VT-3) examination during the next 10-year ISI. Subsequent examinations on the 10-year ISI interval.	100% of accessible baffle-to-former and internal baffle-to-baffle bolt locking devices See Figure 4-2
Primary **Lower Grid Assembly** Alloy X-750 dowel-to-guide block welds	Cracking (SCC), including the detection of separated or missing locking welds, or missing dowels	Initial visual (VT-3) examination no later than two refueling outages from the beginning of the license renewal period. Subsequent examinations on ten-year interval.	100% of accessible locking welds of the 24 dowel-to-guide block welds See Figure 4-4
Primary **Incore Monitoring Instrumentation (IMI) Guide Tube Assembly** IMI guide tube spiders IMI guide tube spider-to-lower grid rib section welds	Cracking (TE/IE), including the detection of fractured or missing spider arms or separation of spider arms from the lower grid rib section at the weld	Initial visual (VT-3) examination no later than two refueling outages from the beginning of the license renewal period. Subsequent examinations on ten-year interval.	100% of accessible top surfaces of 52 spider castings and welds to the adjacent lower grid rib section Figures 4-3 and 4-6
Expansion **Upper Grid Assembly** Alloy X-750 dowel-to-upper fuel assembly support pad welds	Cracking (SCC), including the detection of separated or missing locking welds, or missing dowels	Visual (VT-3) examination	100% of accessible dowel locking welds See Figure 4-6 (i.e., these are similar to the lower fuel assembly support pads)
Expansion **Control Rod Guide Tube Assembly** CRGT spacer castings	Cracking (TE), including the detection of fractured spacers or missing screws.	Visual (VT-3) examination	100% of accessible surfaces at the 4 screw locations (at every 90°) (limited accessibility) See Figure 4-5

Item	Effect (Mechanism)	Examination Method/Frequency	Examination Coverage
Expansion **Lower Grid Assembly** Lower fuel assembly support pad items: pad, pad-to-rib section welds, Alloy X-750 dowel, cap screw, and their locking welds (Note: the pads, dowels, and cap screws are included because of TE/IE of the welds)	Cracking (IE), including the detection of separated or missing welds, missing support pads, dowels, cap screws and locking welds, or misalignment of the support pads	Visual (VT-3) examination	100% of accessible pads, dowels, and cap screws, and associated welds See Figure 4-6
Expansion **Lower Grid Assembly** Alloy X-750 dowel-to-lower fuel assembly support pad welds	Cracking (SCC), including the detection of separated or missing locking welds, or missing dowels	Visual (VT-3) examination	100% of accessible dowels welds See Figure 4-6

Table RAI-2-2 Primary and Expansion Components from CE Designed Plants with Cracking Identified as an Effect and VT-3 Examinations Specified

Item	Effect (Mechanism)	Examination Method/Frequency	Examination Coverage
Primary **Control Element Assembly** Instrument guide tubes	Cracking (SCC, Fatigue) that results in missing supports or separation at the welded joint between the tubes and supports.	Visual (VT-3) examination, no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examination on a ten-year interval. Plant-specific component integrity assessments may be required if degradation is detected and remedial action is needed.	100% of tubes in peripheral CEA shroud assemblies (i.e., those adjacent to the perimeter of the fuel alignment plate). See Figure 4-18
Expansion **Lower Support Structure** Core support column welds	Cracking (SCC, IASCC, Fatigue) including damaged or fractured material	Visual (VT-3) examination, with initial and subsequent examinations based on plant evaluation of SCC susceptibility and demonstration of remaining fatigue life.	Examination coverage determined by plant-specific analysis. See Figures 4-16 and 4-31
Expansion **Control Element Assembly** Remaining instrument guide tubes	Cracking (SCC, Fatigue) that results in missing supports or separation at the welded joint between the tubes and supports.	Visual (VT-3) examination, with initial and subsequent examinations dependent on the results of the instrument guide tubes examinations.	100% of tubes in CEA shroud assemblies. See Figure 4-18

Table RAI-2-3 Primary and Expansion Components from Westinghouse Designed Plants with Cracking Identified as an Effect and VT-3 Examinations Specified

Item	Effect (Mechanism)	Examination Method/Frequency	Examination Coverage
Primary **Baffle-Former Assembly** Baffle-edge bolts	Cracking (IASCC, Fatigue) that results in • Lost or broken locking devices • Failed or missing bolts • Protrusion of bolt heads	Visual (VT-3) examination, with baseline examination between 20 and 40 EFPY and subsequent examinations on a ten-year interval.	Bolts and locking devices on high fluence seams. 100% of components accessible from core side. See Figure 4-23
Primary **Baffle-Former Assembly** Assembly	Distortion (Void Swelling), or Cracking (IASCC) that results in • Abnormal interaction with fuel assemblies • Gaps along high fluence baffle joint • Vertical displacement of baffle plates near high fluence joint • Broken or damaged edge bolt locking systems along high fluence baffle joint.	Visual (VT-3) examination to check for evidence of distortion, with baseline examination between 20 and 40 EFPY and subsequent examinations on a ten-year interval.	Core side surface as indicated. See Figures 4-23, 4-24, 4-25 and 4-26
Primary **Thermal Shield Assembly** Thermal shield flexures	Cracking (Fatigue) or Loss of Material (Wear) that results in thermal shield flexures excessive wear, fracture, or complete separation	Visual (VT-3) no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	100 % of thermal shield flexures. See Figures 4-19 and 4-36
Expansion **Bottom Mounted Instrumentation System** Bottom-mounted instrumentation (BMI) column bodies	Cracking (Fatigue) including the detection of completely fractured column bodies	Visual (VT-3) examination of BMI column bodies as indicated by difficulty of insertion/withdrawal of flux thimbles. Flux thimble insertion/withdrawal to be monitored at each inspection interval.	100 % of BMI column bodies for which difficulty is detected during flux thimble insertion/withdrawal. See Figure 4-35

RAI-3 Eddy current testing (ET) is identified in TR MRP-227 as an inspection method to be used to identify cracking in some PWR RVI components. Clarify whether the acceptance criterion for ET inspections will be based on a "pass – no pass" acceptance criterion (i.e., any ET signals indicating a relevant ET indication would fail the acceptance criterion).

Response: Section 4.2.3 of MRP-227 specifically identifies eddy current surface examination as an electromagnetic testing (ET) method that can be used to supplement visual examination methods, in order to further characterize any detected relevant indications. Therefore, eddy current surface examination is not one of the prime examination methods for which specific examination acceptance criteria are required. When eddy current surface examination is used to supplement visual examination, the purpose will not be to again identify the relevant condition, but instead to further characterize the indication by – for example – confirming the crack-like nature of the indication and more accurately sizing its surface-breaking length. In such a case, the acceptance criteria to be applied will not be *examination* acceptance criteria, but *evaluation* acceptance criteria. These *evaluation* acceptance criteria are referred to in the context of supplementary examinations, engineering evaluations, and repair/replacement in Section 6 of MRP-227. Evaluation acceptance criteria are under development by the PWR Owners Group.

While no specific ET surface examination requirements are provided in the current version of the Inspection Standard (MRP-228), that document does require that all examination techniques, other than visual examination techniques, require a technical justification. The scope of a technical justification is described in MRP-228 in some detail. In addition, since the use of eddy current testing will often be to further characterize the surface-breaking length of a relevant indication detected by visual examination, MRP-228 describes in some detail the uncertainty associated with length sizing of crack-like indications detected by visual examination.

RAI-4 The accessibility of the primary inspection RVI components is not typically addressed. It is therefore not clear how much inspection coverage is necessary to ensure timely detection of aging effects in the primary inspection RVI components. Discuss whether guidance should be provided in TR MRP-227 regarding minimum inspection volumes/areas which must be achieved to take credit for having effectively inspected a particular RVI component.

Response: The intent of MRP-227 is to specify inspections that will identify aging related degradation in a timely fashion. The inspection strategy focuses on components and locations where aging degradation is most likely to occur. However, since there is little evidence that these degradation mechanisms have adversely affected the function of the reactor internals system over the first forty years of reactor operation, complete coverage is not generally required and a sampling strategy is warranted. Nonetheless, MRP-227 requires that any level of degradation found must be thoroughly evaluated both for continued acceptance and for extent of condition (scope expansion).

Examination coverage requirements for the MRP-227 Primary and Expansion inspections are outlined in the rightmost columns of Tables 4-1 through 4-6. The intent of these coverage requirements is to provide an adequate indicator of the type, extent and level of degradation within the plant and the PWR fleet. A general assessment of accessibility

was conducted by the MRP to evaluate the feasibility and adequacy of the proposed inspection requirements. In the few cases where accessibility was severely limited, alternative inspections were required (see response to RAI-14).

Coverage for a visual examination of a single component is defined as the percentage of the target surface area observed in the examination. In some cases, MRP-227 states the coverage requirement as 100% of all accessible surfaces. Consideration of accessibility in terms of target surface is included in the coverage requirements for a number of components as described here:

- B&W Primary and Expansion components are listed in Table RAI-4-1. All of the examinations listed in Table RAI-4-1 are VT-3 examinations designed to determine general component condition. The accessibility survey indicates no severe restrictions exist and normal examination measures will provide adequate coverage.

- The only CE component is the core support barrel. As indicated in Table RAI-4-2 this component appears in three separate Primary and Expansion recommendations. The intent of these inspections is to provide a general monitoring program for active cracking mechanisms. The accessibility survey indicated that normal inspection techniques will provide adequate coverage.

- The core barrel is listed among the Westinghouse components in Table RAI-4-3. This component is similar to the CE core support barrel and it was concluded that normal inspection techniques will provide adequate coverage.

- The remaining Westinghouse components in Table RAI-4-3 are the lower flanges on the control rod guide tube assemblies and the lower support column bodies. Both of these components are parts of redundant systems where a single flaw will not result in failure. The intent of both examinations is to provide a reasonable sampling of the component condition. Normal inspection techniques should provide adequate sampling for the components in Table RAI-4-3.

The MRP-227 recommendations also refer to accessibility as a factor in determining coverage for inspection of a variety of bolts, locking devices and lock welds. These recommendations span all three reactor designs and include both UT volumetric and VT-3 visual examinations. In these cases, MRP-227 uses the term coverage to refer to the fraction of components inspected rather than the coverage associated with any particular examination. The question of coverage for UT volumetric inspections of bolts is addressed as part of the technical justification for the inspection technique.

The requirement to inspect accessible bolts in primary components is generally based on the observation that the overwhelming majority of bolts in the system should be readily accessible. Alternative recommendations were provided for systems of bolts where accessibility was identified as a significant concern. Because there is redundancy built into the bolted system, failure of an isolated bolt or bolts does not threaten the continued safe operation of the system.

MRP-228 requires that the coverage be reported with the examination results.

The effectiveness and timeliness of this approach at the individual plant level is further augmented by the aggregation of all fleet inspection results (see MRP-227 Section 7.6), which will identify unexpected degradation results and upon review would lead to revised inspection guidance if necessary. An experience-based evolution of PWR internals inspection and evaluation guidelines has always been considered a necessary part of the process in that results of early examinations will provide the basis for adjusting the requirements. The industry will provide PWR internals inspection summaries to the MRP including extent of examination and coverage issues if they arise. The combined data will provide the industry with a large inspection sample size that will ensure timely identification of emerging degradation issues as well as insights into unanticipated inspection coverage limitation. If unanticipated evidence of active degradation is identified in the combined data set, or if actual inspection coverage is unexpectedly and excessively limited, the inspection requirements will be adapted.

The coverage requirements in MRP-227 are adequate to ensure timely detection of aging effects in the reactor vessel components. Setting of quantitative coverage requirements might lead to either extraordinary efforts for minor gains in coverage to meet artificial minimums or less than complete examinations for plants that easily exceed the minimum requirements. Further definition of minimum inspection coverage in terms of volumes or areas for particular reactor vessel internals components would not significantly improve MRP-227.

Table RAI-4-1 B&W Primary and Expansion Components with Coverage Requirement Stated in Term of Accessible Surface Area

Item	Effect (Mechanism)	Examination Method/Frequency	Examination Coverage
Primary **Core Support Shield Assembly** CSS cast outlet nozzles	Cracking (TE), including the detection of surface irregularities, such as damaged or fractured material	Visual (VT-3) examination during the next 10-year ISI. Subsequent examinations on the 10-year ISI interval.	100% of accessible surfaces. See Figure 4-9
Primary **Core Support Shield Assembly** CSS vent valve discs (Note 1)			100% of accessible surfaces. (See BAW-2248A, page 4.3 and Table 4-1.) See Figures 4-10 and 4-11
Primary **Core Support Shield Assembly** CSS vent valve top retaining ring CSS vent valve bottom retaining ring CSS vent valve disc shaft or hinge pin (Note 1)	Cracking (TE), including the detection of surface irregularities, such as damaged, fractured material, or missing items	Visual (VT-3) examination during the next 10-year ISI. Subsequent examinations on the 10-year ISI interval.	100% of accessible surfaces (See BAW-2248A, page 4.3 and Table 4-1.) See Figures 4-10 and 4-11
Primary **Core Barrel Assembly** Baffle plates	Cracking (IE), including the detection of readily detectable cracking in the baffle plates	Visual (VT-3) examination during the next 10-year ISI. Subsequent examinations on the 10-year ISI interval.	100% of the accessible surface within 1 inch around each flow and bolt hole See Figure 4-2
Primary **Incore Monitoring Instrumentation (IMI) Guide Tube Assembly** IMI guide tube spiders IMI guide tube spider-to-lower grid rib section welds	Cracking (TE/IE), including the detection of fractured or missing spider arms or separation of spider arms from the lower grid rib section at the weld	Initial visual (VT-3) examination no later than two refueling outages from the beginning of the license renewal period. Subsequent examinations on ten-year interval.	100% of accessible top surfaces of 52 spider castings and welds to the adjacent lower grid rib section Figures 4-3 and 4-6
Expansion **Control Rod Guide Tube Assembly** CRGT spacer castings	Cracking (TE), including the detection of fractured spacers or missing screws.	Visual (VT-3) examination	100% of accessible surfaces at the 4 screw locations (at every 90°) (limited accessibility) See Figure 4-5

Table RAI-4-2 CE Primary and Expansion Components with Coverage Requirement Stated in Term of Accessible Surface Area

Item	Effect (Mechanism)	Examination Method/Frequency	Examination Coverage
Primary **Core Support Barrel Assembly** Upper (core support barrel) flange weld	Cracking (SCC)	Enhanced visual (EVT-1) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	100% of the accessible surfaces of the upper flange weld. See Figure 4-15
Expansion **Core Support Barrel Assembly** Lower core barrel flange	Cracking (SCC, Fatigue)	Enhanced visual (EVT-1) examination, with initial and subsequent examinations dependent on the results of the upper (core support barrel) flange weld examinations.	100 % of accessible welds and adjacent base metal. See Figure 4-15
Expansion **Core Support Barrel Assembly** Remaining core barrel assembly welds	Cracking (SCC)	Enhanced visual (EVT-1) examination, with initial and subsequent examinations dependent on the results of core barrel assembly upper flange weld examinations.	100 % of one side of the accessible weld and adjacent base metal surfaces for the weld with the highest calculated operating stress. See Figure 4-15

Table RAI-4-3 Westinghouse Primary and Expansion Components with Coverage Requirement Stated in Term of Accessible Surface Area

Item	Effect (Mechanism)	Examination Method/Frequency	Examination Coverage
Primary **Control Rod Guide Tube Assembly** Lower flange welds	Cracking (SCC, Fatigue)	Enhanced visual (EVT-1) examination to determine the presence of crack-like surface flaws in flange welds no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	100 % of outer (accessible) CRGT lower flange weld surfaces and adjacent base metal. See Figure 4-21
Primary **Core Barrel Assembly** Upper core barrel flange weld	Cracking (SCC)	Periodic enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	100 % of one side of the accessible surfaces of the selected weld and adjacent base metal. See Figure 4-22
Expansion **Core Barrel Assembly** Core barrel flange, core barrel outlet nozzles, Lower core barrel flange weld	Cracking (SCC, Fatigue)	Enhanced visual (EVT-1) examination, with initial examination and re-examination frequency dependent on the examination results for upper core barrel flange.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal See Figure 4-22
Expansion **Lower Support Assembly** Lower support column bodies (non cast)	Cracking (IASCC)	Enhanced visual (EVT-1) examination, with initial examination and re-examination frequency dependent on the examination results for Upper core barrel flange weld,	100 % of accessible surfaces See Figure 4-34
Expansion **Lower Support Assembly** Lower support column bodies (cast)	Cracking (IASCC) including the detection of fractured support columns	Visual (EVT-1) examination.	100% of accessible support columns See Figure 4-34

RAI-5 During the extended period of operation, some PWR RVI components are subject to high levels of neutron radiation which may lead to irradiation embrittlement and a loss of fracture toughness and the potential for irradiation-assisted stress corrosion cracking. In combination, these effects may lead to the potential for component failure under some design basis loading conditions. Explain how licensees will be expected to account for potential reduction in fracture toughness when evaluating cracks that are detected during the required inspections, in particular when establishing the frequency of subsequent inspections after cracking is identified.

Response: Cracking detected during examinations will be evaluated using the evaluation acceptance criteria and methodologies currently being developed by the PWROG Materials Subcommittee.

Section 6 of MRP-227 describes potential evaluation steps that can be followed when cracking is detected during the specified examinations, whether due to the loss of fracture toughness or from irradiation-assisted stress corrosion cracking. These steps include the assessment of cracking through limit load and/or fracture mechanics evaluations, depending upon the extent of the loss of fracture toughness. The steps also include the potential for increasing the frequency of subsequent examinations as determined from the evaluations, using crack growth rates described in Section 6.2.4. The descriptions in Section 6 of MRP-227 are "For Information Only." Other information on evaluation of detected cracking is available from other sources, such as "Fracture Toughness Of Irradiated Stainless Steel In Nuclear Power Systems," S. Fyfitch, et al., 14th International Conference on Environmental Degradation of Materials in Nuclear Power Systems, Virginia Beach, Virginia, August 23-27, 2009 (to be published). This reference contains a Figure 2 with a lower bound curve of the currently available data for the effect of fluence on fracture toughness of austenitic stainless steel materials. The saturated fracture toughness value of 38 MPa$\sqrt{m}$ can be used for fluences greater than 15 dpa.

RAI-6 Loose parts could be generated due to deterioration of some PWR RVI components during the extended period of operation. Provide information which addresses how the following consequences of loose parts generation were considered in development of the inspection program given in TR MRP-227. (a) potential for fuel bundle flow blockage and consequential fuel damage, (b) potential for interference with control rod operation, and (c) potential for impact damage on reactor internals.

Response: In general, loose parts are included in existing plant-specific monitoring and evaluation procedures and they remains a plant specific issue. As the intent of the inspection program developed in MRP-227 is to discover potential age-related degradation, all RV internals items were evaluated for aging degradation, and consequences of the generation of loose parts were considered. However, the results and consequences of loose parts generation were previously considered, evaluated, and documented in Section 11 of MRP-156 and Section 6 of MRP-157 and noted throughout the Failure Modes, Effects, and Criticality Analysis (FMECA) efforts summarized in MRP-190 and MRP-191. Specific information for items (a)-(c) above is available in these reports.

As an example, below the core, the reactor internals consist of the support structure and instrumentation assemblies. All of these components are very substantial and would not be expected to be damaged by loose parts. The primary purposes of these components are

to position and support the core and direct the flow to the fuel assemblies. Large loose parts that are capable of being lifted by the flow will be filtered by these components and likely be lodged in or pinned against one of these structures. Smaller parts or fragments that can pass through the flow holes are typically trapped in the lower end of the fuel assembly. Flow area blockage associated with loose parts in the lower reactor internals will have an insignificant effect on core performance since the flow will be redistributed downstream of the blockage and in the lower span of the fuel assemblies.

RAI-7 Alloy 600 PWR RVI components and their associated welds manufactured from Alloys 82 and 182 are susceptible to primary water stress corrosion cracking (PWSCC) when exposed to PWR reactor coolant water. In Table 3-1 of TR MRP-227, the following Babcock and Wilcox (B&W) Alloy X-750 PWR RVI components were welded with Alloy 82 material and yet they were classified under "N" category which excludes inspections for these PWR RVI components: (1) dowel-to-core barrel cylinder welds, (2) dowel-to-upper grid rib section bottom flange welds, (3) dowel locking welds, (4) dowel-to-guide block welds, and (5) dowel-to-distributor flange welds. Even though stress levels in these components may not exceed the threshold levels, the NRC staff considers it to be likely that PWSCC can potentially occur due to the introduction of cold work during fabrication. In light of this observation, provide an explanation for excluding inspection requirements for these B&W PWR RVI components.

Response: As explained in Section 2.6 of MRP-231, these dowel and dowel locking welds were used during assembly of the B&W design RV internals and although they may be susceptible to PWSCC, they no longer have an RV internals function after assembly. Thus, they were re- classified as No Additional Measures.

RAI-8 When exposed to a light-water reactor temperatures of approximately 500 °F or higher, the 17-4 precipitation hardened (PH) martensitic stainless steel (MSS) that has previously been subjected to aging (heat treatment) at about 1100 °F can experience thermal embrittlement and an increase in hardness and a reduction in Charpy V-notch impact test toughness. Operating experience from Oconee Nuclear Station (Information Notice (IN) 2007-02, ADAMS Accession Number ML070100459) shows that thermally embrittled 17-4 PH MSS is susceptible to failure when exposed to unexpected loading conditions. In IN 2007-02, the NRC staff recommended that licensees prevent the deleterious effects of thermal embrittlement in the 17-4 PH MSS components by identifying aging degradation (i.e., cracks), implementing early corrective actions, and monitoring and trending age-related degradation. Therefore, the NRC staff requests that the TR MRP-227 report should include thermal embrittlement as an aging effect for any 17-4 PH MSS RVI components.

Response: The initial screening, as performed in accordance with MRP-175, includes thermal aging embrittlement as a potential degradation concern for all martensitic PH stainless steels (see Table 3-2). As noted in MRP-156 and MRP-157, no RV internals component items were fabricated from Type 17-4 PH materials in the B&W, CE, or Westinghouse designs. However, there are two component items in the B&W design that were fabricated from Type 15-5 PH martensitic stainless steel (vent valve top and bottom retaining rings). These have been evaluated in MRP-231 (see Table 3-8) and are included in the Primary component item examinations (see Table 4-1 of MRP-227).

RAI-9 With respect to the management of cast austenitic stainless steel (CASS) aging and embrittlement TR MRP-227 does not appear to address the program's compliance with the requirements specified in the relevant Generic Aging Lessons Learned (GALL) Report AMPs. Provide a discussion of how TR MRP-227 adequately addresses the requirements specified in GALL AMP, XI.M12, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)," and GALL AMP XI.M13, "Thermal Aging and Neutron Embrittlement of Cast Austenitic Stainless Steel (CASS)," for CASS materials used in PWR RVI components. Alternatively, if the management CASS PWR RVI component aging is not treated within the scope of TR MRP-227, provide a proposed modification of the report which documents how licensees are expected to manage this mechanism outside of the TR MRP-227 program.

Response: The inspection requirements for CASS component items in MRP-227 provides augmented inspection to detect cracking in components that have potentially experienced a loss of fracture toughness. The screening and FMECA processes conducted to develop the MRP-227 inspections considered the implications of loss of fracture toughness in the limited number of reactor internals component items that are potentially fabricated from cast austenitic stainless steel and provides relevant aging management recommendations. The inspections are based on an item-by-item evaluation on a generic vendor design basis. This process defines an adequate Aging Management Program consistent with the intent of GALL AMP XI.M12 and X1.M13. Based on the findings of this study it is our belief that implementation of the MRP-227 Guidelines provides appropriate aging management for irradiated cast stainless steel. It is our recommendation that GALL AMP XI.M13 be withdrawn to allow establishment of requirements for aging management cast stainless steel internals in GALL AMP XI.M16.

GALL AMP XI.M12 defines the program required to manage thermal embrittlement in cast stainless steel reactor components. The MRP-227 process followed the screening process provided in GALL AMP XI.M12 to identify components potentially subject to thermal embrittlement. Nearly all known cast austenitic stainless steel items were screened-in for thermal embrittlement due to the fact that the chemical composition had not been reviewed at the time. The only exception was the cast austenitic stainless steel vent valve body in the B&W design whose ferrite content had already been determined to be below the screening criteria using Hull's equivalent factors in NUREG/CR-4513 Rev. 1.

All screened-in CASS items went through the FMECA review and evaluation process. Some CASS items in the Westinghouse designed PWRs were reclassified as "No Additional Measures" through the FMECA process. All remaining screened-in CASS items are in the final Primary or Expansion groups in MRP-227. However, if the ferrite content can be determined to be below the screening threshold during any plant-specific reviews, those items can be removed from the Primary or Expansion groups with appropriate technical justification.

GALL AMP XI.M13 contains all of the essential parts of the current XI.M12 for CASS thermal embrittlement and adds an additional fluence threshold of 1E17 n/cm^2, E>1 MeV. The fluence used in screening for irradiation embrittled CASS items in MRP-175 was 6.7E20 n/cm^2, E>1 MeV. However, the reduction in the fluence threshold had relatively little impact on the evaluation because the reduction in fracture toughness due to thermal embrittlement was already identified as a potential aging effect. In addition to being screened in for thermal embrittlement, many of the cast austenitic stainless steel items were also screened at the higher threshold for irradiation embrittlement. At high

neutron fluence the loss of fracture toughness due to irradiation will bound the fracture toughness loss in any CASS item due to thermal embrittlement.

It is important to note that the MRP-227 recommendations are not based on detailed flaw evaluations. Any component that would require a flaw evaluation to demonstrate acceptability for continued service was included in the aging management program. Determination of the fracture toughness value to be used in the evaluation of a flaw location with a fluence between 1E17 and 6.7E20 n/cm^2, E>1 MeV might require evaluation of potential interactions between thermal and irradiation embrittlement mechanisms. However, this type of detailed analysis was not used in the basis for the inspection recommendations in MRP-227.

The CASS items most susceptible to thermal and irradiation embrittlement are placed in the Primary group and their supplemental examinations during the 10-year ISI program suggested by X1.M13 have been provided in MRP-227. The remaining CASS items are placed in the Expansion group whose inspection requirement, if triggered, will be based on the findings from the Primary CASS items. This process incorporates all of the GALL AMP XI.M13 concerns within the MRP-227 recommendations. The general reasoning behind the MRP-227 inspections is outlined below.

The MRP-227 I&E Guidelines are based on the component item specific screening, categorization, analysis and strategy development processes detailed in MRP-134, MRP-175, MRP-190, MRP-191, MRP-231 and MRP-232. Although the basis for the specific recommendations may vary from item-to-item, they are generally based on a combination of the following considerations:

1. Many of the cast austenitic stainless steels items placed in the Expansion group are not part of the core support structure. Cracking or failure of these items has no direct effect on the safety or function of the plant. Therefore, inspection strategies for them do not require the same level of inspection as that required for primary pressure boundary or core support structures.

2. Although it is difficult to demonstrate that the peak load in any reactor internal CASS item is less than 5 ksi, cast austenitic stainless steels were not used for large load bearing applications. Loads in these items tend to be low and compressive. Active cracking mechanisms such as IASCC and SCC have not been observed in PWR primary system CASS components. Therefore there is a low likelihood of crack initiation in PWR CASS components.

3. Many of the cast austenitic stainless steel items are parts in a redundant system (e.g., core support columns or CRGT spacers). Failure of a single item will not lead to failure of the system.

4. In many cases, the design specifications for these component items allowed, but did not require, the use of cast austenitic stainless steels. A small and as yet undetermined number of units actually used cast materials in these component items. Because there was no generic analysis indicating the type or composition of the material used to fabricate these component items, they were all screened in for thermal embrittlement.

5. The treatment of cast stainless steel reactor internals aging is treated within the scope of MRP-227. Potential aging degradation of the cast components due to thermal and irradiation embrittlement was thoroughly considered in the development of the MRP-227 inspection guidelines. Incorporation of the MRP-

RAI-10 According to Section A.1.4 in MRP-175, "Materials Reliability Program: PWR Internal Aging Degradation Mechanism Screening Threshold Values," susceptibility to SCC in nickel-based Alloy X-750 PWR RVI components depends on the type of heat treatment that is performed on the alloy. High temperature heat treatment processes that are used on Alloy X- 750 components offer better resistance to SCC than the other age hardened heat treatment processes. Licensee determination of the heat treatment applied to their Alloy X-750 PWR RVI components would appear to be a critical parameter in ensuring the licensee's AMP will adequately manage the potential effects of aging. Discuss whether this determination should be included as a license renewal application action item.

> **Response:** MRP-227 does not explicitly require determination of the heat treatment for X-750. But licensee determination of the heat treatment applied to the Alloy X-750 RV internals items is performed by current industry practice and does not need to be included as an MRP-227 licensee action item, but should remain a license renewal application action item. The implementation of the guidelines in MRP-227 is governed by the Materials Guidelines Implementation Protocol (Addendum D) of NEI 03-08.

RAI-11 Following on to RAI-10, additional aspects of the TR MRP-227 methodology may need to be addressed by license renewal applicant action items for applications currently under review or those that have yet to be submitted to the NRC. The NRC staff requests the MRP's assistance in identifying potential action items which are: (1) necessary to provide plant-specific information to complete the AMP; (2) necessary to confirm applicant compliance with important assumptions underlying the MRP-227 methodology; or (3) other considerations.

> **Response:** With respect to the first potential action item, guidance on regulatory submittals is outside the scope of MRP-227; however the MRP is willing to work with the NRC to help as requested.
>
> With respect to the second potential action item, Section 2.4 of MRP-227 states explicitly that
>
> > *"The guidelines are intended to serve as the primary basis for owner preparation of a reactor internals AMP in accordance with the requirement cited in Section 7. It is beyond the scope of the guidelines, however, to ensure the satisfaction of every plant-specific license renewal or power uprate commitment. Plant-specific commitments remain the responsibility of the owner."*
>
> Licensee action items for a typical plant will relate to guidelines development assumptions. Section 2.4 of MRP-227 contains a list of the assumptions that need to be verified as applicable by individual plant owners. These assumptions are cited here for completeness.
>
> > *"The guidelines are based on a broad set of assumptions about plant operation, which encompass the range of current plant conditions for the U.S. domestic fleet of PWRs. The functionality analyses and supporting aging management*

strategies in MRP-231 [13] and MRP-232 [14] provide the basis for these guidelines. These evaluations were based on representative configurations and operational histories, which were generally conservative, but not necessarily bounding in every parameter.

General assumptions used in the analysis include:

- *30 years of operation with high leakage core loading patterns (fresh fuel assemblies loaded in peripheral locations) followed by implementation of a low-leakage fuel management strategy for the remaining 30 years of operation;*
- *base load operation, i.e., typically operates at fixed power levels and does not usually vary power on a calendar or load demand schedule; and*
- *no design changes beyond those identified in general industry guidance or recommended by the original vendors*

These assumptions are a conservative representation of U.S. PWR operating plants, all of which implemented low leakage core loading patterns early in operating life. The recommendations are thus applicable to all U.S. PWR operating plants as of May 2007 for the three designs identified.

These guidelines are also considered applicable to plants that have replaced components or component assemblies; however, alternatives can be technically justified. Plant modifications made or considered after this date should be reviewed to assess impacts on strategies contained in these guidelines."

Therefore, all of the important assumptions underlying the MRP-227 methodology were aggregated into Section 2.4 of the document.

RAI-12 was clarified by an NRC letter dated 09/28/2009.
Provide the loading sources that were used in determining the peak stress values for each Pressurized Water Reactor (PWR) Reactor Vessel Internal (RVI) component. Loading sources may include pressure, thermal, deadweight, residual stress (e.g., from fabrication/installation, welding), hydrodynamic, preload, and other sources that contribute to normal loading. Identify which if any of these loading sources produce cyclic or transitory stresses. Transitory loading source may include, for example, mechanical, thermal, hydrodynamic, or pressure transient. Also, indicate the portion of the peak stresses which is due to static loading sources and the portion attributed to cyclic or transitory load sources that may contribute to fatigue. The NRC staff believes that plants that have been implementing power uprates will have to assess whether the peak stress values for any given PWR RVI component are affected by power uprate conditions to determine if their plant is bounded by the assumptions underlying TR MRP-227.

Response: MRP-227 provides a robust set of inspection recommendations based on a combination of pre-existing evaluations, expert elicitation, complex interactive modeling, and analysis. The MRP-227 recommendations are based on projected aged component conditions occurring as a result of normal power operation, which are controlled by multiple factors including stress. These projections assume normal loading conditions. Transitory conditions encountered in normal operation were also considered in projecting fatigue responses. The inspection recommendations target vulnerable locations based on representative plant analysis with the goal of preemptively detecting any potential degradation in a component prior to affecting its ability to perform its intended function

for all design load conditions. Re-qualification of peak loads is a design requirement for the plant uprate analysis, further evaluation to demonstrate compliance with MRP-227 is not necessary.

The initial screening and categorization process elicited opinions from experts in the field of reactor internals design and analysis. The experts were instructed to consider all factors that could lead to degradation by any of the eight aging related degradation mechanisms. The screening evaluation of SCC or IASCC initiation considered all normal operating loading conditions including fabrication/installation loads. Transitory loading sources were a factor considered by the experts in the evaluation of fatigue. Stress relaxation was considered as a potential degradation mechanism in components where pre-loads were applied in installation. Stress was not a primary factor in the evaluation of void swelling, irradiation embrittlement, thermal embrittlement or wear. The expert panel was encouraged to draw on data from stress reports and plant uprate evaluations. However, no original analysis to demonstrate design qualification was included in this portion of the study.

Functionality analyses using detailed finite element models were then developed for the most highly irradiated reactor internals components. Models of representative plants were analyzed for each of the three original PWR vendors. Due to the unique challenges associated with evaluating the complex interactions between the irradiation driven aging degradation mechanisms, the effects of irradiation on the stress-strain behavior of the steels, stress relaxation/creep and void swelling were incorporated in the finite element models. Specifically, 40 fuel cycle, sixty-year duration finite element analyses were conducted to subject selected reactor internals assemblies to time dependent temperature and dose distributions based on normal 100% power operation and a standard startup/shutdown transient. The assembly materials were allowed to deform from elastic-plastic behavior, as well as creep, stress relaxation, and void swelling. IASCC and loss of preload was monitored to allow for the determination of susceptible components.

The finite element models for each representative internals configuration were subjected to normal operation and heat up and cool down thermal gradients and pressure loading for forty 18 month fuel cycles. Core loading considering dose and heat generation were also included. Prior to fuel cycle loading, applicable mechanical preloads were applied to bolts and tie rods. Specific to the Combustion Engineering finite element model, weld residual stresses were imposed onto the full penetration welds retaining the baffle plates to the former plates.

These analyses were conducted using a nonlinear aging material module for 304 and 316 stainless steels. The material module incorporated time and radiation dependent embrittlement effects in the stress-strain behavior. Further, the module allowed the simulations to impose irradiation creep, stress relaxation, and void swelling behavior once thresholds (if any) for each were reached. As a result, these mechanisms could allow preloads on bolts and tie rods to increase or decrease. IASCC susceptibility was monitored throughout these analyses. However, no failure was imposed on any component found to be susceptible.

Functional evaluation of the remaining, non-irradiation degradation mechanisms (Wear, SCC, Thermal Embrittlement and Fatigue) was based primarily on operating experience and engineering judgment rendered by design and analysis experts. Again, the subject experts were encouraged to refer to existing analysis including both stress reports and

plant uprate studies. Relevant loading conditions for these non-irradiation degradation mechanisms correspond to those considered in a typical plant uprate evaluation.

In summary, a wide range of loading conditions was included in the evaluations performed to assess components subject to age related damage mechanisms. Analyses against ASME Code rules for qualification and acceptability was not needed to support identification of components and locations for inspection and monitoring programs. The recommendations are based on lessons learned from representative plant analyses and should not be considered to be a bounding calculation. All plant uprate programs require a rigorous qualification analysis to demonstrate that the original design requirements are not violated. It is unlikely that power uprates would affect the conclusions of MRP-227. Applicability criteria for the guidelines are provided in Section 2.4 of MRP-227. There would be no value added to the process by requiring additional plant-specific analysis to demonstrate that their plant is bounded by the assumptions underlying the MRP-227 report.

RAI-13 Certain degradation mechanisms (e.g., void swelling in B&W PWR RVI components) are not inspected for in a particular reactor type. Why does the program not require the most susceptible location for each mechanism in each reactor-type (i.e., B&W, Combustion Engineering, or Westinghouse) be inspected as a primary component to insure that each degradation mechanism is not occurring within the reactor?

Response: The intent of the MRP-227 inspection strategy is to support comprehensive aging management programs for both the individual units and the entire fleet. All eight aging degradation mechanisms identified for the reactor internals are addressed by these recommendations. The strategy employed for each mechanism is outlined in the response to RAI-14.

Some of the aging degradation mechanisms are not directly observable in non-destructive examinations. In this case, the relevant inspections detect the consequences of the degradation mechanism. The functionality analysis, which linked void swelling and irradiation-induced stress relaxation to the stresses that control both IASCC and fatigue, was instrumental in this process.

For example, there are no "Primary" RV internals component items for void swelling in the B&W designed units in MRP-227 (Rev. 0). This is the only aging degradation mechanism in the B&W units that does not have a "Primary" or "Expansion" component item. However, there are primary inspection items that are impacted by the effects of void swelling. These effects are expected to be the first observable consequences of void swelling.

The functionality analysis of aging in the highly irradiated core barrel assembly demonstrates the complex interaction between aging mechanisms in the reactor internals. The B&W former plates were initially classified as Category C for void swelling while the baffle plates, baffle-to-former bolts, and baffle-to-baffle bolts were classified as Category B for void swelling. Subsequent functionality analysis (MRP-229) of the B&W core barrel assembly, based on finite element analysis, showed the level of void swelling in the core barrel assembly to be insignificant. The highest local void swelling after 60 years is calculated to be approximately 3 % and confined to very small regions at the reentrant corners of the baffle plates. Volumetric void swelling for most areas in the core

barrel assembly is predicted to be below one percent. For example, the predicted 3 % void swelling in the reentrant corner of baffle plates will not be of any concern for embrittlement and coolant flow. In contrast, similar analysis by Westinghouse predicted higher maximum void swelling in the Westinghouse designed baffle-former assembly due to higher temperatures.

However, the functionality analysis of the B&W core barrel assembly (MRP-229) did indicate that small levels of void swelling can induce significant deformation discontinuities in the core barrel assembly and prying of baffle-to-former bolts at certain elevations with relatively low level fluence. As a result, overloading has been identified as a potential effect for the baffle-to-former bolts and their locking devices.

Therefore, there are four reasons why there are no "Primary" components for void swelling in the B&W units in MRP-227:

- The maximum localized void swelling during a 60-year lifetime is predicted to be approximately 3% and is below 1% for most regions. Therefore, localized void swelling is of no concern.
- The low level of void swelling could increase IASCC susceptibility and potential overload for some baffle-to-former bolts and locking devices, which are already identified as "Primary" component items by MRP-227.
- The locations of the baffle-to-former bolts and locking devices most affected by the void swelling are not among the highest void swelling locations. Therefore, the most susceptible (highest) void swelling component items (locations) are not among the first to demonstrate the effects of void swelling.
- The Westinghouse units will lead the B&W units due to the higher temperatures.

Direct monitoring of the most susceptible location is not necessarily the most effective means of detecting aging degradation in the reactor internals. Although direct inspection for void swelling is not specifically included in the MRP-227 recommendations, void swelling is effectively monitored by inspecting for these consequential effects.

RAI-14 Discuss how the PWR RVI components in each reactor design considered to be the most susceptible to (or most likely to first demonstrate the effects of) a particular degradation mechanism did, or did not, get binned in the primary inspection component group for that design.

Response: As stated in RAI-13, the intent of the MRP-227 inspection strategy is to support comprehensive aging management programs for both the individual units and the entire fleet. Each component and degradation mechanism was reviewed as part of the comprehensive program. All eight degradation mechanisms identified degradation mechanisms are managed in the context of the MRP-227 recommendations. With the exception of void swelling in the B&W design (as discussed in the response to RAI-13) all eight degradation mechanisms are represented in the Primary inspections.

The final classification as Primary or Expansion component items was based on a variety of factors including relative susceptibility, severity of consequence and component accessibility. The susceptibility to aging degradation was ranked as part of the FMECA process (see response to RAI-22). However, not all of the most susceptible component items are placed in the MRP-227 "Primary" inspection group. Some are placed instead in the "Expansion" inspection group due to the following two considerations:

- Safety consequence consideration – Some of the most susceptible component items have no or lower safety consequence compared to the "Primary" component items, which are equally susceptible to the degradation mechanism.
- Accessibility consideration – Some of the most susceptible component items are inaccessible. However, the "Primary" inspection group contains at least one equally susceptible, but accessible component to support managing the degradation mechanism.

Details of this process are contained in MRP-231 and MRP-232.

Each of the "Primary" component items identified in MRP-227 is among the most susceptible to or most likely to first demonstrate the effects of the age-related degradation mechanisms. There are eight aging-related degradation mechanisms covered by the MRP-227 recommendations. These mechanisms are:

1. Irradiation-Assisted Stress Corrosion Cracking (IASCC)
2. Irradiation Embrittlement
3. Void Swelling
4. Irradiation-Enhanced Stress Relaxation/Creep
5. Stress Corrosion Cracking (SCC)
6. Fatigue
7. Wear
8. Thermal Aging Embrittlement

Note that these eight aging-related degradation mechanisms include both aging effects that potentially could lead to failure (cracking and wear) and aging effects that lead to changes in material properties (embrittlement) that are not directly observable using standard NDE techniques. The inspection strategy and aging management approach varies depending on the degradation mechanism.

<u>IASCC, Irradiation Embrittlement, Void Swelling and Irradiation Stress Relaxation/Creep</u> - The effects of these aging degradation mechanisms were evaluated from the functionality analysis results. Based on this analysis, it is possible to identify the peak location for each of these mechanisms. However, the first observable effects of these mechanisms may not occur at the peak location. In particular, void swelling and irradiation-enhanced stress relaxation are critical factors in determining the stress distribution in the irradiated structure. The first observable effects of these are expected to occur where the stresses indicate a potential for IASCC or fatigue cracking. Direct observation of these mechanisms is neither practical nor required to manage aging. As discussed in the response to RAI-13, this is why the B&W units have no "Primary" or "Expansion" components linked directly to void swelling. The MRP-227 guidelines provide an integrated approach to these aging degradation mechanisms. There are no field-proven inspection methods to directly monitor fracture toughness, but these locations are monitored for cracking. Any flaw tolerance analysis would assume limiting fracture toughness values for irradiated material.

<u>SCC</u> - Although SCC of the 300 series stainless steel alloys is thought to be an unlikely failure mechanism MRP-227 does require inspection of key core barrel welds or bolting to

monitor this effect. The "Primary" components were selected based on a combination of susceptibility (high stress), severity (critical to core support) and accessibility.

Fatigue – Fatigue concerns were identified in the analysis primarily on the basis of available cumulative usage factors (CUF). MRP-227 explicitly recognizes the need for fatigue evaluations in several components and requires inspection only if acceptable fatigue life cannot be demonstrated by analysis. Primary and Expansion recommendations for fatigue were based on a combination of severity (high CUF) and severity of consequence. In some highly redundant components, the strategy was based on sampling of accessible components.

Wear – Each potential wear location represents a unique combination of conditions. There is no basis for ranking the susceptibility of wear locations. The MRP-227 inspection strategy for wear is generally to require a visual inspection. If visual evidence of wear is reported, supplemental exams may be necessary to characterize the wear.

Thermal Embrittlement - The MRP-227 inspection strategy for managing thermal embrittlement of cast austenitic stainless steels is discussed extensively in the response to RAI-9. There is no basis for predicting the component with the highest susceptibility to the combined effects of thermal and irradiation embrittlement. If the fracture toughness of a cast austenitic stainless steel component is potentially reduced and there is a reasonable possibility of a pre-existing flaw or an active cracking mechanism, inspections to identify crack-like defects were required. There are no inspections required to monitor fracture toughness. Any flaw tolerance analysis would assume limiting fracture toughness values for embrittled material.

RAI-15 This RAI has been withdrawn by the NRC Staff.

RAI-16 Clarify the conditions under which design basis event (DBE) effects on component performance were considered. How does this approach provide reasonable assurance that the margins against failure are adequately maintained during the license renewal period?

Response: The consideration of DBE effects had minimal impact on the Inspection and Evaluation (I&E) Guidelines provided in MRP-227. The original screening and categorization of internals components was based solely on evaluation of potential age-related degradation under normal operating conditions. Inclusion of components in the I&E Guidelines was dependent on susceptibility to one or more of the eight age-related degradation mechanisms, and as such did not need to consider DBE. Later in the process, during the Failure Modes, Effects, and Criticality Analysis (FMECA), the impact of aging on the ability of a component to meet performance requirements under DBE effects was taken into consideration to determine the potential severity of failure consequences. An example of this type of consideration is given in Section 4.1.2.1 of MRP-232, where the report states that a panel of experts reviewed various stress reports and calculation notes (which included the full range of DBE events), in order to identify the worst-case stress levels in the various core barrel components, eventually leading to the selection of the upper core support barrel flange as the Primary component location. This type of consideration was not universal, nor did it need to be. However, knowledge of the full range of loading conditions was considered in order to make appropriate aging management strategy recommendations.

Subsequently, the vendor functionality analyses were based on nominal operating conditions – 30 years of high leakage core loading, followed by 30 years of low leakage core loading, with the intent of determining the effects of combinations of various age-

related degradation effects. The full range of DBE effects was not used in these functionality calculations. However, when age-related degradation effects are detected during the examinations specified in MRP-227, the suitability of the degraded component for continued service will necessarily take into consideration the full range of DBE effects. This is discussed in Section 6 of MRP-227, which is "For Information Only." This section also discusses in some detail the loading conditions that will be considered in the engineering evaluations.

RAI-17 Component failure due to the same degradation mechanism is not considered to be a common cause failure because of the expectation that damage initiation and growth occurs at different times. However, certain DBEs could potentially lead to a plant condition (damage state) that would not occur unless multiple components were degraded. Discuss how the potential for multi-component failure due to a DBE was considered as part of the development of the MRP-227 program.

Response: Analysis of failure, whether single component or multi-component, was not part of the process used to identify the MRP-227 inspection requirements. As stated in the response to RAI-16, design-basis loading conditions were not considered during the initial screening and categorization of internals components. Only the susceptibility of the components to long-term age-related degradation effects was considered. However, the potential for multi-component failure from similar levels of degradation in system components was considered by the subject matter experts during the susceptibility assessments in the failure modes, effects, and criticality analysis (FMECA) portion of the program. The "improbable" failure of similar components due to the same age-related degradation mechanism was noted in the Susceptibility column of the FMECA (in MRP-190). The severity of consequence was not evaluated for these; however, when the consequence was deemed severe, the Susceptibility metric was bolded in the FMECA table. An example is the round bars (in the Core Support Shield Assembly in B&W plants), where the potential for similar states of stress corrosion cracking (SCC) of the round bars was identified. This was discussed in Section 3.4.1 of MRP-190. In addition to identifying improbable common cause failures, the FMECA also noted cascading (or dependent) failures as part of the severity of consequence assessment. This was discussed in Section 3.4.2 of MRP-190.

During the functionality analysis portion of the program, the potential for similar stages of degradation in various components of an assembly was also taken into account. The recommendations provided a systematic approach to understanding degradation in these systems. Therefore the examination requirements included the possibility of examination of more than one location in an assembly, or 100% of the accessible locations in an assembly, or several locations in different assemblies subject to similar states of degradation.

The potential for component failure as the result of similar states of *detected* degradation effects is within the purview of supplementary examination (to determine more accurately the actual state of degradation) or engineering evaluation (to consider whether simultaneous or progressive failure might result from a combination of similar degradation states and design-basis loadings). As noted in the response to RAI-16, these items were beyond the scope of the MRP-227 recommendations. Multiple component failures would be considered as part of the engineering evaluation. One example would

be the minimum bolting patterns developed to determine the allowable patterns of failed bolts in the Westinghouse baffle-former structure.

RAI-18 Clarify how plant-specific differences were considered within the FMECA. Discuss whether any additional plant-specific analyses are required, either as a supplement to TR MRP-227 or as identified plant-specific action items, in order to assure that FMECA analysis supporting the TR MRP-227 program is applicable to a given facility.

Response: During the development of the FMECA, a conservative approach was taken. Differences in internals design or operation were considered and factored into the expert panel's qualitative assessments for determining potential susceptibility of degradation and reduced capacity to perform intended functions. Where critical to final categorization of components, these considerations are captured in the text of the documents.

MRP-190 for Babcock & Wilcox (B&W) design and MRP-191 for Combustion Engineering and Westinghouse designs provided a detailed discussion of how design variances were considered. The failure modes, effects, and criticality analysis (FMECA) approach used was not a quantitative probabilistic risk analysis, but rather a semi-quantitative approach with expert elicitation. The experts were instructed to consider how design and operational differences affected the evaluation. MRP-190 Section 5 and MRP-191 Section 6 provide description of the categorization process using the FMECA for the designs considered in MRP-227.

The applicability of the evaluations in MRP-190 and MRP-191 to specific plants was noted in the text of the documents. In addition, Section 4 of MRP-190 stated that the assumptions "are either bounding or methodological, and do not require plant-specific verification for each of the B&W-designed units." While a similar statement was not contained in MRP-191, the intention is the same and the applicability of the outcome of the process to the operating fleet was documented in the listing of plants considered and process of evaluation especially noting the grouping approach to addressing the design variances. The inspection requirements included for individual components, noting specifically Combustion Engineering designs for bolted or welded configurations, were contained in MRP-227.

Section 1 of MRP-227 stated that the contents were applicable to the currently operating pressurized water reactors (PWRs) as of the date of publication. The demonstration of the applicability of MRP-227 to individual units was specified in Section 2.4. Design changes that may have occurred subsequent to May 2007 are managed by the plant configuration control process.

RAI-19 Discuss how a licensee will demonstrate adherence to the reference core loading pattern on a unit-specific basis. Address plant-to-plant variability in neutron flux at various peripheral core locations. Confirm, based on significant operating experience, that "low leakage" core designs, when normalized by power density, have peripheral neutron fluxes that are consistently within the estimates for the generically studied plants.

Response: The core loading patterns used in the MRP-227 reference documents were chosen to represent known operating practice, they are not intended to be used as a reference for plant-specific analysis. The intention of using the representative core

loading patterns was not to bracket operation, but to perform an analysis that demonstrates both historic and current fuel management programs. The MRP-227 inspection recommendations based on these calculations are robust and do not require the utility to perform additional analysis of core loading patterns to qualify their applicability.

The condition of the internals at the time of the first required inspections is dominated by the power distribution used to represent the first thirty years of full power operation. During this period the analysis assumed that the fresh fuel was loaded in the peripheral fuel assemblies. This "out-in" loading pattern produced results in relatively high heat loadings and neutron fluences in the near core structure. In practice all plants in the United States abandoned fuel management based on the "out-in" loading prior to thirty years of operation. There are no current or planned fuel management programs that would result in more deleterious conditions than those assumed in this analysis during the first thirty years of operation. For this reason there is no reason to require any plant to perform an analysis to demonstrate adherence to the assumed core loading pattern prior to performing the first round of inspections. The timing and extent of the first round of MRP-227 examinations is governed by damage that has already been accumulated.

The representative power distributions used for the simulation of years 31 to 60 incorporate the effects of aggressive power uprate programs. Qualification of the core loading pattern is considered in the design analysis for the plant uprate. Although it is not possible to anticipate all possible future options, both current fuel management practice, which maximizes fuel utilization, and concerns about neutron damage in the reactor pressure vessel preclude return to the practice of loading fresh fuel in the periphery locations. It is unlikely that future core loading patterns would invalidate the assumptions of the analysis.

Although the shift from "out-in" core loading patterns to low-leakage patterns resulted in a sharp decrease in the peak temperature in the internals structure, the shift had minimal effect on the location of the peak temperature or the character of the peak damage. There is no reason to expect that changing the loading pattern would change the base inspection recommendations. The MRP-227 recommendations are based on reasonable assumptions about the effects of power uprates. In many cases power uprates can be accomplished without significantly increasing the heat or neutron loading to the internals. Return to the more aggressive core loading patterns could conceivably result in a decrease in the re-inspection interval. However, there is no reason to anticipate any change of this scale.

MRP-227 is intended to be a living document. The MRP will monitor both inspection results and plant operating experience and make appropriate modifications. There is currently no need to require plants to demonstrate adherence to any reference core loading practice.

RAI-20 Provide a technical basis to justify the examination acceptance criteria, the sufficiency and relevancy of the links between primary and expansion group components (why were those particular links chosen), and the expansion criteria. Discuss also the technical basis that applied to place certain components in the primary category while others were placed in the expansion category.

Response: The RAI will be treated as three separate items – first, to provide the technical basis for the examination acceptance criteria; second, to provide the technical basis for the links between particular primary and expansion components; and third, to provide the technical basis for the expansion criteria. The response to the second item will address the technical basis for placing one component in the Primary group while a related component is placed in the Expansion group.

Item 1 – Examination Acceptance Criteria. MRP-227 contains three types of examination acceptance criteria. For visual examination (and surface examination as an alternative to visual examination), the examination acceptance criterion is the absence of any of the prescribed relevant conditions. For volumetric examination, the examination acceptance criterion is the capability for reliable detection of indications in bolting, as demonstrated in the examination technical justification, which then leads to a pass/fail determination. For physical measurements, the examination acceptance criterion for the acceptable tolerance in the measured differential height from the top of the plenum rib pads to the vessel seating surface in B&W plants has been generically established and is given in Table 5-1, while the Westinghouse plant internals hold-down spring height limit will be established on a plant-specific basis.

The use of visual examination relevant conditions (and the absence of relevant conditions as a visual examination acceptance criterion) is consistent with the ASME Code Section XI rules for visual examination. MRP-227 has taken the use of visual examination relevant conditions to a higher level by providing descriptions that are more specific to components and degradation effects, so that the absence of these specific degradation effect conditions gives improved confidence in the examination results.

The technical basis for volumetric examination relevant conditions can be found in MRP-228 and its supporting documentation, where the review of existing bolting ultrasonic examination technical justifications has demonstrated the indication detection capability of at least two vendors, and where vendor technical justification is a requirement prior to any additional bolting examinations.

The technical basis for the generic physical measurement of the differential height from the top of the plenum rib pads to the vessel seating surface in B&W plants is given in MRP-231 (Materials Management Program: Aging Management Strategies for B&W PWR Internals). The technical basis for the plant-specific measurement of Westinghouse plant internals hold-down spring height will be provided on a plant-specific basis.

Item 2 – Primary Component Links to Expansion Components. The technical basis for the links between primary components and related expansion components is found in the supporting documents to MRP-227, in particular MRP-231 and MRP-232. MRP-232 provides a very formal process, referred to as the waterfall method, for aggregating groups of components together with common characteristics and then identifying the component (or components) that should receive maximum attention, with other components receiving lesser attention until the waterfall process indicates that greater attention is needed. Section 2.4 of MRP-232 describes this formal process well. MRP-231 describes a similar approach in a much less formal way on Page 1-2, with the following key words:

> "The new categorization combines the results of functionality assessment with item accessibility, operating experience, existing evaluations, and prior

examination results to determine the appropriate program elements for maintaining the long-term functions of PWR internals safely and economically. Therefore, AREVA NP has revised the preliminary categorization (Categories A, B, and C), based on the recent evaluation results and subsequently has re-categorized them as Primary, Expansion, Existing, and No Additional Measures."

An illustrative example explains the process. Core shroud bolts were found to be the most affected components for IASCC and fatigue in Combustion Engineering plants through the waterfall process, requiring volumetric (UT) examination. Other CE plant bolting systems, such as barrel-shroud bolts, guide lug insert bolts, and core support column bolts, are less affected by the environment, resulting in their classification as Expansion components. This grouping is an example of an inspection waterfall because of the common requirement for UT examination to detect cracking from either IASCC or fatigue.

Item 3 – Expansion Criteria. The expansion criteria were determined through the use of expert panels. The panel consisted of a diverse group of experts who jointly evaluated existing information and eventually reach agreement on the criteria for expanding the group of primary components when sufficient evidence of unexpected degradation effects are detected. The process and technical bases are described in "Letter to Reactor Internals Focus Group from MRP, Subject: *Minutes of the Expert Panel Meetings on Expansion Criteria for Reactor Internals I&E Guidelines*, MRP 2008-036 (via email), June 12, 2008)".

An expert panel worksheet was prepared for each Expansion component and its linked Primary component. The first block in the worksheet provided the name of the Expansion component, a description of that component (including a reference to any relevant drawings in MRP-227), the degradation mechanism (effect), and the proposed examination method/coverage. The second block in the worksheet provided much the same information on the linked Primary component, but added such information as the timing of initial and periodic examinations. The third block in the worksheet provided a comparison of the key parameters between the two component locations – such as material, temperature, and irradiation dose. The fourth block in the worksheet identified additional considerations, such as operating experience and design grouping considerations (welded versus bolted designs). Finally, the fifth block captured the expert panel deliberation and eventual recommendation. This last block also captured any interim issues that required action or further study prior to a final recommendation.

An illustrative example is provided by the baffle-to-former bolts (Primary) and the barrel-former and lower support column bolts (both Expansion) in Westinghouse plants (see Table 5-3, page 5-17), where the criteria are quite specific about the combination of fluence and the number of unacceptable baffle-to-former bolts.

RAI-21 Many of the acceptance criteria provided in TR MRP-227 are vague such as finding "detectable crack-like surface indications," or "damaged or fractured material," or "readily detectable cracking." It's not clear that these criteria will be uniformly interpreted or implemented from plant to plant. Discuss the need to develop more detailed acceptance criteria on a plant-specific basis and how will the sufficiency of these criteria be established.

Response: In determining what level of specificity to include in MRP-227, it was anticipated that visual examiners would need guidance for recording relevant conditions but not be so exacting that other potential relevant conditions would go unrecorded. Even so, MRP-227 went beyond the general condition relevant condition descriptions found in ASME Code Section XI by being much more specific about the types of discontinuities that would be expected from particular degradation effects for particular components. Thus the intended relevant condition is targeted by MRP-227.

Additionally, these targeted descriptions are not for final acceptance of the component item other than by their complete absence. Thus the mere detection of such crack-like conditions requires further disposition through the corrective action process.

To provide uniform implementation of examination acceptance criteria and interpretation of relevant conditions visual inspection personnel are required by MRP-228 Section 2.3.4 to receive a minimum of 4 hours of training on inspection requirements and specific information related to the component:

a. A review of inspection video recordings of the specific components showing the types of flaws or relevant conditions to which the components are susceptible.
b. Various types of non-relevant indications that may be encountered.
c. Identification of areas prone to cracking, including details and characteristics of cracks that might be found in these areas.
d. The effects of surface conditions on detecting and evaluating indications.

Utilities are also required to conduct site-specific training for all personnel that will evaluate the visual inspection data prior to inspections for each refueling outage. The training will include pertinent information such as prior inspection results, utility specific procedural requirements (such as acceptance criteria) and other applicable topics.

Besides visual examination, ultrasonic (UT) examination is also required by MRP-227 for the detection of cracks in bolting. For this examination technique and any other technique chosen other than visual examination, the PWR Internals Inspection Standard, MRP-228, requires a uniform approach for NDE system qualification. This requirement for the NDE system qualification is contained in Section 2.1 of MRP-228 and involves the development of a written technical justification providing a detailed explanation of the examination process to be employed. This process is specifically designed to provide uniformity and consistency in the performance of bolting UT examinations.

RAI-22 The screening criteria groups materials into susceptibility levels for each degradation mechanism: highly susceptible, moderately susceptible, susceptible, and "below the screening criteria." Discuss the criteria used to distinguish among the different levels of susceptibility.

Response: In MRP-190, susceptibility (likelihood that an age-related degradation mechanism (ARDM) might occur) was defined as follows:

- A – Improbable: not likely to occur
- B – Unexpected: not very likely to occur, though possible; conditions are such that the age-related degradation mechanism is not expected to occur very often
- C – Infrequent: likely to occur, conditions are such that the age-related degradation mechanism is expected to occur occasionally

- D – Anticipated: very likely to occur; conditions are such that the age-related degradation mechanism is expected to occur

In MRP-191, susceptibility was similarly assessed with a slight variance in terminology from MRP-190 and a review of operating history to affirm categorization. MRP-191 Table 6-2, Component Failure Likelihood, defined the four categories as:

- None – Expert panel concurs that failure of the component is not credible in a 60-year lifetime (i.e., no screened-in age-related degradation mechanisms or other evidence to support a concern). No known failures.
- Low – Expert panel believes the component is unlikely to fail in a 60-year lifetime either due to known or potentially emerging issues based on current knowledge base. No known failures.
- Medium – Expert panel believes there is the potential for concern, multiple degradation modes are a possibility, or believes further investigation is merited to solidify classification. No known failures.
- High - Expert panel expects this component to fail or cannot exclude the possibility of failure or susceptibility to failure within the 60-year lifetime. Known failures.

For each component item in the failure modes, effects, and criticality analysis (FMECA) tables, a susceptibility to degradation was assigned as A, B, C, or D, or None, Low, Medium, or High. This assignment (or distinguishing among the different levels of susceptibility) was performed by the expert panel based on their experience and expertise (using the definitions noted) and is a qualitative assessment. The expert panel process is discussed in Section 3.2 of MRP-190 and in Section 6 of MRP-191.

RAI-23 Discuss whether an evaluation was performed for any specific high consequence of failure PWR RVI components such that their inspection might be warranted even in the absence of a currently identifiable mechanism. Are there any PWR RVI components that should be monitored through in-service inspection to protect against unforeseen failure due to the emergence of a potential future degradation mechanism?

Response: The Failure Modes, Effects and Criticality Analysis (FMECA) that was conducted during the screening and categorization of RV internals component items independently considered both the susceptibility of component items to age-related degradation and the consequences of potential component failure.

The weighting of susceptibility and consequences (see, e.g., Figure 4-1 in MRP-189, Reference 8) are such that it is possible to have components with a high safety consequences ranking and a very low susceptibility ranking (with very low susceptibility just above the screening value). All components with these characteristics for the B&W design were placed initially within Category C, implying that additional measures for detection and monitoring of age-related degradation effects could be needed. It should be pointed out that a Category C initial ranking for a component did not necessarily lead to a Primary or Expansion final group. However, the process did allow for components with high safety significance and low susceptibility to be included in Category C. In fact, no RV internals item in the B&W units had received a safety ranking of this level during the FMECA process. Therefore, there are no items in the B&W units that would be considered of such high safety significance to require inservice inspection beyond the current ASME inservice inspection requirements if it is not currently considered to be

susceptible to any age-related degradation. Westinghouse followed a similar process in MRP-191 of independently reviewing components and placing them in categories.

For components that were originally placed in Category A, which satisfied of all screening criteria, no attempt was made to formally re-rank such components on the basis of very high safety significance. However, the categorization decision for each of these Category A components was reviewed by the expert panels at the beginning of the FMECA evaluation, in order to assure that no information had been overlooked that could alter that decision. One of the pieces of information considered in that process was the potential for very high safety consequences. Therefore, it is unlikely that a non-susceptible component would be subject to very high safety consequences.

It should be pointed out that components with significant safety consequences of failure are also core support components. Therefore, the current ASME Section XI inspections of core support structures are considered to be sufficient to address any unforeseen failure due to the emergence of a potential future degradation mechanism in components not specifically placed in one of the MRP-227 inspection groups. ASME Section XI inspections considered sufficient to manage aging in areas with known or potential degradation were placed in the Existing Programs MRP-227 group.

RAI-24 Relevant US and international operating experience with respect to RVI components is not summarized. It is important to indicate what prior RVI component inspections have identified, in particular with respect to justifying the adequacy of existing programs and as part of the basis for the examination requirements (e.g., type, periodicity, importance) identified in MRP-227.

Response: The RAI observation is correct. Relevant U.S. and international operating experience with respect to PWR internals was not summarized either in the body of MRP-227 or in an appendix. Relevant operating experience was taken into account in the FMECA assessments documented in MRP-189, MRP-190, and MRP-191, and relevant operating experience is documented both formally and informally in the vendor aging management strategy reports (MRP-231 and MRP-232) that led to the recommendations in MRP-227. For example, Section 3.2.2.1 (Industry IASCC Experience and Existing Evaluations) of MRP-231 provides a formal, in-depth discussion of an important segment of that operating experience. On the other hand, Section 4.1.6 (In-Core Instrumentation) of MRP-232 provides an embedded discussion of operating experience with respect to ICI thimble tubes. In many cases throughout MRP-231 and MRP-232, the relevant operating experience discussion may only involve a single summary sentence.

As a way of capturing in one convenient location, the operating experience contained in MRP-231 and MRP-232 could be included in the GALL Report Chapter XI.M16 revision.

RAI-25 The cumulative usage factor values for several B&W components need to be confirmed during a comprehensive search of all existing stress and fatigue calculations for the PWR internals. Discuss how such items are intended to translate into plant-specific action items.

Response: Because of the lack of specific ASME design rules for core support structures at the time of design and construction, Section III of the ASME Code was used as a guideline for the design criteria for the B&W-design PWR internals. The qualification of

the B&W-design 177-FA PWR internals was accomplished by both analytical and test methods.

The FMECA expert panel concluded that fatigue usage for all components and items in the operating 177-FA B&W-designed PWRs would be low. Therefore, the three non-bolting component items (the plenum cylinder, the upper grid rib section, and the core barrel cylinder) initially screened-in for fatigue, were reclassified to Category A for fatigue in MRP-189. All bolting component items screened-in for fatigue as a result of stress relaxation were unaffected regardless of their CUF values.
There are no current plans for MRP to confirm the CUF values for the non-bolting items since this is a concern specific to the B&W-design units. For example, TLAA fatigue evaluations, performed as part of License Renewal for the Oconee units for replacement structural bolting and extension of the FIV evaluations listed in BAW-10051 to the period of extended operation, are documented in NUREG-1723. This has confirmed that there are no additional structural analyses that would change the reclassification of the three non-bolting items listed above. AREVA will work with the remaining B&W-design licensees on a plant-specific basis to provide confirmation of low fatigue usage for these non-bolting items, similar to what was completed for Oconee.

RAI-26 The implications of void swelling are indicated as "dimensional change and distortion…" and it is also noted that "severe void swelling may result in cracking under stress." However, it is not indicated that void swelling can lead to reduced fracture toughness in materials even though it is noted in Section 3.2.7 of TR MRP-227 that "severe swelling (>5%) has been correlated with extremely low fracture toughness values." It is not clear how much void swelling is needed before distortion is detectable via VT-3 examination in susceptible PWR RVI components and whether this threshold for dectectability will also address the concern over potential loss of fracture toughness due to void swelling. Provide a discussion of this topic.

Response: The functionality analysis indicated that there were limited regions in the Westinghouse baffle-former structure and some CE core shroud designs that may experience swelling greater than 5% by volume. These regions are potentially subject to the additional loss of toughness effect associated with severe void swelling. This swelling is predicted at the peak temperature locations at the baffle-former plate intersections in the Westinghouse designs and in the central flanges in the CE core shrouds. Due to the lower predicted temperatures in the B&W baffle structure, the predicted peak swelling levels are < 5%. It should also be noted that functionality analysis of the structural bolts in the Westinghouse, CE and B&W designs predicted swelling well below this 5% limit. Therefore, there is no concern about this swelling related embrittlement mechanism in the structural bolting.

Analysis results for the Westinghouse and CE designed plants are summarized in MRP-230. The peak temperatures in these near-core locations occur away from the cooled surfaces and are generally limited to the plate interior (or interfaces). The volume potentially subject to these severe embrittlement conditions is a small fraction of the entire core structure. In addition, the assumption that the plants operated for thirty years with fresh fuel loaded in the peripheral core positions ("Out-in core loading") led to sufficiently conservative assumption that the high temperatures associated with this loading pattern were present for the first thirty years of reactor operation. In more realistic conditions, the volume of material exceeding 5% would be even smaller than that predicted in MRP-230.

The combination of VT-1, EVT-1 and VT-3 visual inspections recommended for the Westinghouse core baffle structure and the CE core shroud structure are designed to provide the information required to manage degradation due to void swelling, embrittlement and IASCC. The visual inspections are targeted at the locations where displacement or separation of plates is most likely to be noted. The extent and character of the distortion at these locations is discussed in MRP-230. These inspections are included to provide validation of the swelling calculations. If distortion at these locations is not observed, it is reasonable to assume that the MRP-230 analysis continues to bound the behavior of the structure. Inspection for IASCC cracks is required at locations where swelling leads to stresses above the threshold for cracking. The most likely locations for cracking are generally removed from the peak swelling locations. In the analysis of any observed flaw, the effect of irradiation embrittlement on the fracture toughness of the surrounding material would have to be considered. This would include any possible effects of severe void swelling.

September 28, 2009

Mr. Christian B. Larsen
Nuclear Vice President & Chief Officer
Electric Power Research Institute
3420 Hillview Avenue
Palo Alto, CA 94304-1338

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION RE: ELECTRIC POWER RESEARCH INSTITUTE TOPICAL REPORT 1006596, "MATERIALS RELIABILITY PROGRAM (MRP): PRESSURIZED WATER REACTOR INTERNALS INSPECTION AND EVALUATION GUIDELINES (MRP-227 – REV. 0) (TAC NO. ME0680)

Dear Mr. Larsen:

By letter dated January 12, 2009, Electric Power Research Institute (EPRI) submitted for U.S. Nuclear Regulatory Commission (NRC) staff review Topical Report (TR) 1006596, "Materials Reliability Program (MRP): Pressurized Water Reactor Internals Inspection and Evaluation Guidelines." Upon review of the information provided, the NRC staff has determined that additional information (RAI) is needed to capture the initial set of technical questions related to the NRC staff's review of TR MRP-227 to support completion of the review. The NRC staff will however issue another set of RAIs based on its review of the MRP-227 report and its supporting reports. By e-mail dated September 22, 2009, Anne Deema, Senior Project Manager, and I agreed that the NRC staff will receive your response to the enclosed RAI questions within 30 days of issuance of this letter.

In addition, by letter dated August 24, 2009, the NRC requested that EPRI respond to a set of NRC RAIs. The NRC RAIs are available in the Agencywide Documents Access and Management System, Accession No. ML092250603. During a September 10, 2009, conference call with EPRI representatives to discuss these RAIs, the NRC staff decided to provide a written clarification to RAI #12 and withdraw RAI #15 from consideration. The revised RAI #12 is enclosed. By this letter, the NRC staff is no longer requesting a written response to RAI #15.

If you have any questions regarding the enclosed RAI questions, please contact me at 301-415-3610.

Sincerely,

/RA/ by SRosenberg for

Tanya M. Mensah, Senior Project Manager
Special Projects Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project Nos. 669 and 689

Enclosure:
RAI questions

cc w/encl: See next page

If you have any questions regarding the enclosed RAI questions, please contact me at 301-415-3610.

Sincerely,

Tanya M. Mensah, Senior Project Manager
Special Projects Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project Nos. 669 and 689

Enclosure:
RAI questions

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ADAMS ACCESSION NO.: ML092650329 **NRR-106**

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REQUEST FOR ADDITIONAL INFORMATION (RAI)

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT (TR) 1006596, "MATERIALS RELIABILITY PROGRAM (MRP): PRESSURIZED

WATER REACTOR INTERNALS INSPECTION AND EVALUATION GUIDELINES

(MRP-227 – REV. 0)

ELECTRIC POWER RESEARCH INSTITUTE

PROJECT NO. 669

1. The NRC staff requests that the following Electric Power Research Institute (EPRI) documents be submitted expeditiously to the NRC to support the staff's review of Topical Report (TR) Materials Reliability Program (MRP)-227.

   a) MRP-211, "Materials Reliability Program: PWR Internals Age-Related Material Properties, Degradation Mechanisms, Models, and Basis Data - State of Knowledge."

   b) MRP-229, "Materials Reliability Program: Functionality Analysis for B&W Representative PWR Internals."

   c) MRP-230, "Materials Reliability Program: Functionality Analysis for Westinghouse and Combustion Engineering Representative PWR Internals."

2. By letter dated August 24, 2009, the NRC requested that EPRI respond to a set of NRC Requests for Additional Information (RAIs). The NRC RAIs are available in the Agencywide Documents Access and Management System, Accession No. ML092250603. During a September 10, 2009, conference call with EPRI representatives to discuss these RAIs, the NRC staff decided to provide a revision to RAI #12 in writing, and to withdraw RAI #15 from consideration, as shown below:

   RAI-12
   Provide the loading sources that were used in determining the peak stress values for each Pressurized Water Reactor (PWR) Reactor Vessel Internal (RVI) component. Loading sources may include pressure, thermal, deadweight, residual stress (e.g., from fabrication/installation, welding), hydrodynamic, preload, and other sources that contribute to normal loading. Identify which if any of these loading sources produce cyclic or transitory stresses. Transitory loading source may include, for example, mechanical, thermal, hydrodynamic, or pressure transient. Also, indicate the portion of the peak stresses which is due to static loading sources and the portion attributed to cyclic or transitory load sources that may contribute to fatigue.

   The NRC staff believes that plants that have been implementing power uprates will have to assess whether the peak stress values for any given PWR RVI component are affected by

power uprate conditions to determine if their plant is bounded by the assumptions underlying TR MRP-227.

RAI-15 (Withdrawn by the NRC Staff)

(via email)

November 11, 2009

Ms. Tanya M. Mensah
Senior Project Manager
Special Projects Branch, Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Subject: REQUEST FOR ADDITIONAL INFORMATION RE: ELECTRIC POWER RESEARCH INSTITUTE TOPICAL REPORT 1016596,'MATERIALS RELIABILITY PROGRAM: PRESSURIZED WATER REACTOR INTERNALS INSPECTION AND EVALUATION GUIDELINES (MRP-227-REV. 0)' (TAC NO. ME0680), September 28, 2009 (please note the corrected EPRI Product Number – 1016596 – from that in the actual RAI request letter – 1006596)

Reference:

1. Letter Tanya Mensah (NRC) to Christian B. Larsen (EPRI), same subject, dated September 28, 2009
2. Letter and Affidavit, Tuan Nguyen (EPRI) to NRC Document Control Desk, Request for Withholding Commercial Documents, dated November 12, 2009

Dear Ms. Mensah:

In response to your September 28 letter (Reference 1) requesting that EPRI provide copies of reports to support NRC review of EPRI Report 1016596, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-Rev. 0)" we are forwarding eight copies of the following two documents:

1) *Materials Reliability Program: Functionality Analysis for B&W Representative PWR Internals (MRP-229-Rev. 1).* EPRI, Palo Alto, CA: 2009. 1019090.
2) *Materials Reliability Program: Functionality Analysis for Westinghouse and Combustion Engineering Representative PWR Internals (MRP-230-Rev. 1).* EPRI, Palo Alto, CA: 2009. 1019091

These documents have been forwarded to the Document Control Desk by Reference 2 (copy attached) requesting that this copyrighted information be withheld from public disclosure.

As discussed during the conference call on November 2, 2009, our purpose in providing the functionality analysis reports is to support responding to questions on the guidelines and clarify the overall process used in their development. We are not asking for a review of the functionality analysis reports. The aging management strategies developed for the guidelines are based on: operating experience and prior examinations results, prior evaluations, component functionality

and inspectability, and engineering judgment and expert opinion. The functionality analysis was used to provide further insights for inspection locations and to confirm expert opinions.

Again, as discussed in our November 2 conference call, we have recently completed our verification of the materials model underlying the functionality analyses and are now in the process of re-running the those analyses. We do not expect these re-runs to have an impact the aging management strategies or the associate guidelines. When these re-runs are complete, we will update both MRP-229 and MRP-230 and should be able to provide Revision 2 of the documents to you in March or April of 2010.

If you have any questions, please contact Christine King at 650-855-2605, or Anne Demma at 650-855-2026.

Best Regards,

Dennis P. Weakland

Dennis Weakland
First Energy
Chairman, Materials Reliability Program

Cc: Victoria Anderson (NEI)
Jeff Ewin (INPO)
Tuan Nguyen (EPRI)
Christine King (EPRI)
Anne Demma (EPRI)

NEI Project Nos. 669 and 689

November 12, 2009

Mr. Christian B. Larsen
Nuclear Vice President & Chief Officer
Electric Power Research Institute
3420 Hillview Avenue
Palo Alto, CA 94304-1338

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION RE: ELECTRIC POWER RESEARCH INSTITUTE TOPICAL REPORT 1016596, "MATERIALS RELIABILITY PROGRAM (MRP): PRESSURIZED WATER REACTOR INTERNALS INSPECTION AND EVALUATION GUIDELINES (MRP-227 – REV. 0)" (TAC NO. ME0680)

Dear Mr. Larsen:

By letter dated January 12, 2009, Electric Power Research Institute submitted for U.S. Nuclear Regulatory Commission (NRC) staff review Topical Report 1016596, "Materials Reliability Program (MRP): Pressurized Water Reactor Internals Inspection and Evaluation Guidelines." Upon review of the information provided, the NRC staff has determined that additional information (RAI) is needed to support completion of the review. The NRC staff will however issue another set of RAIs based on its review of the MRP-227 report and its supporting reports. By e-mail dated November 9, 2009, Anne Deema, Senior Project Manager, and I agreed that the NRC staff will receive your response to the enclosed RAI questions within 30 days of issuance of this letter.

If you have any questions regarding the enclosed RAI questions, please contact me at 301-415-3610.

Sincerely,

/RA/

Tanya M. Mensah, Senior Project Manager
Special Projects Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project Nos. 669 and 689

Enclosures:
1. RAI Questions
2. RAI Attachment Containing Tables 1, 2, and 3

cc w/encl: See next page

Mr. Christian B. Larsen
Nuclear Vice President & Chief Officer
Electric Power Research Institute
3420 Hillview Avenue
Palo Alto, CA 94304-1338

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION RE: ELECTRIC POWER RESEARCH INSTITUTE TOPICAL REPORT 1016596, "MATERIALS RELIABILITY PROGRAM (MRP): PRESSURIZED WATER REACTOR INTERNALS INSPECTION AND EVALUATION GUIDELINES (MRP-227 – REV. 0)" (TAC NO. ME0680)

Dear Mr. Larsen:

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If you have any questions regarding the enclosed RAI questions, please contact me at 301-415-3610.

Sincerely,

/RA/
Tanya M. Mensah, Senior Project Manager
Special Projects Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project Nos. 669 and 689

Enclosures:
3. RAI Questions
4. RAI Attachment Containing Tables 1, 2, and 3

cc w/encl: See next page

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DATE	11/12/2009	11/12/2009	11/12/2009

OFFICIAL RECORD COPY

Nuclear Energy Institute
Electric Power Research Institute

Project No. 689
Project No. 669

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REQUEST FOR ADDITIONAL INFORMATION (RAI)

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT (TR) 1016596, "MATERIALS RELIABILITY PROGRAM (MRP):

PRESSURIZED WATER REACTOR INTERNALS INSPECTION

AND EVALUATION GUIDELINES

(MRP-227 – REV. 0)

ELECTRIC POWER RESEARCH INSTITUTE (EPRI)

PROJECT NO. 669

In a letter dated January 12, 2009, EPRI submitted a TR MRP-227, Rev. 0, "Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines," which addresses an aging management program (AMP) for the PWR reactor vessel internal (RVI) components. On July 2, 2009, EPRI provided additional reports that support the technical bases used for developing the AMP and these reports were submitted to the NRC staff for information only. The NRC staff is in the process of reviewing TR MRP-227, Rev. 0, and the supporting reports. Based on the review conducted thus far, the NRC staff has developed the following RAI. The NRC staff, however, expects to issue additional sets of RAI questions based on its review of TR MRP-227, Rev. 0, and its supporting reports.

RAI-1

It is not evident to the NRC staff whether or not the inspection and evaluation (I&E) methodology in TR MRP-227-Rev. 0, or in other MRP report methodologies that support the TR MRP-227, Rev. 0, methodology, are:

a) in compliance with the scoping, screening, and aging management requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 54, or;
b) in conformance with the NRC staff's aging management recommendations in NUREG-1800, Revision 1, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," (SRP-LR), Section A.1, "Aging Management Review – Generic (Branch Technical Position (BTP) RLSB-1)."[1]

Perform a comparison of the recommended I&E methodology in TR MRP-227, Rev. 0, and its supporting MRP aging management methodologies to the NRC staff's recommended aging effect identification and management guidance in SRP-LR BTP RLSB-1. Demonstrate the compliance of the TR MRP-227, Revision 0, methodology with the requirements of 10 CFR Part 54 and with the NRC staff's recommendations in SRP-LR BTP RLSB-1. In addition, justify any

[1] Henceforth, the reference to this Branch Position in this set of RAIs will be abbreviated as SRP-LR BTP RLSB-1.

non-compliances of your proposed methodology with the stated requirements in 10 CFR Part 54 or non-conformances with the recommendations in SRP-LR BTP RLSB-1.

RAI-2

TR MRP-227, Rev. 0, does not make reference to Nuclear Energy Institute (NEI) Report No. NEI 95-10, Revision 6,[2] as an applicable industry license renewal report; nor does TR MRP-227, Rev. 0, provide any discussion on how it will be used to conform to the recommendations in NEI 95-10, Rev. 6, when implementation of the I&E recommendations are applied as license renewal activities and methods for aging management of RVI components

Discuss how and provide the basis for why the recommended I&E scoping/screening process, inspection activities, evaluation methods and acceptance criteria in TR MRP-227, Rev. 0, are considered to be in conformance with the recommendations in NEI 95-10, Rev. 6, paying particular attention to conformance with Chapters 2, 3, and 4 of the NEI 95-10, Rev. 6, and the license renewal processes identified therein for implementation in Figures 2.0-1, 3.0-1, 4.1-1, 4.2-1, and 4.3-2.

RAI-3

By letter dated February 20, 2009, the NRC granted a fee waiver under the provisions of 10 CFR 170.11(a)(1)(iii) to support a non-fee billable review of TR MRP-227, Rev. 0. The fee waiver was granted by the NRC on the basis that the methodology of the report would be used to update the NRC staff's aging management review (AMR) items for PWR RVI components, as given in the following NRC license renewal guidance documents: (1) the SRP-LR; and (2) NUREG-1801, "Generic Aging Lessons Learned Report" (GALL), Volumes 1 and 2.

The NRC's recommended AMR items in Tables of the GALL Report, Revision 1, Volume 1 (henceforth referred to as Table 1 AMR items) are given in the following AMR column format:

Table X. Summary of Aging Management Programs for the .... System Evaluated in Chapter IV of the GALL Report							
ID	Type	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Required	Related Generic Item	Unique Item

2 NEI 95-10, Revision 6, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54-The License Renewal Rule." NEI 95-10, Rev. 6, has been endorsed for use in Regulatory Guide 1.188, "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses"[Sept. 2005].

The NRC's recommended AMR items in Tables of the GALL Report, Revision 1, Volume 2 (henceforth referred to as Table 2 AMR items) are given in the following AMR column format:

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM B2 (or B3; B4) Reactor Vessel Internals (PWR) – Westinghouse (Combustion Engineering; Babcock and Wilcox)							
Item	Link	Structure and/or Component	Material	Environment	Aging Effect/Mechanism Required	Aging Management Program (AMP)	Further Evaluation

These AMR item formats have been adopted for use in Tables 3.0-1 and 3.0-2 of NEI 95-10, Revision 6, and endorsed for use in Regulatory Guide 1.188, Revision 1.

Provide Table 1 AMR items for Westinghouse (W), Combustion Engineering (CE), and Babcock & Wilcox (B&W) RVI component commodity groups that are in conformance with the guidelines for formatting Table 1 AMR items in Figure 3.0-1 of NEI 95-10, Revision 6. The NRC staff also requests that you provide the Table 2 AMR items for W, CE, and B&W RVI component commodity groups that are in conformance with the guidelines for formatting Table 1 AMR items in Figure 3.0-2 of NEI 95-10, Revision 6.

Note—Tables 1, 2 and 3 shown in Enclosure 2 to this document provide information related to the aging effects which were not included for the RVI components listed in Tables 4-1 through 4-6 of TR MRP-227, Rev. 0.

RAI-4

Provide a new draft GALL AMP XI.M16, "PWR Vessel Internals Program" in a format that conforms to the recommended program element criteria in SRP-LR BTP RLSB Section A.1.2.3 and that can be adopted for the contents of an applicant's PWR Vessel Internals Program when the license renewal application is submitted to the NRC for NRC staff approval.

RAI-5

Provide the basis for why the functionality analysis (FA) in Figure 2-2 was applied to Category C RVI components and not to Category B RVI components.

RAI-6

Clarify whether or not the existing the methodology in TR MRP-227, Rev. 0, can be applied to a PWR facility whose reactor core loading pattern operating history is not bounded by the assumptions in the report.

If the methodology can be applied, justify why that is the case. If the methodology cannot be applied to these PWRs, identify what actions a licensee with a non-conforming PWR would have

to take in order to develop a plant-specific AMP for its RVI components, which is consistent with the intent of TR MRP-227, Rev. 0.

Identify whether license renewal applicants should demonstrate that their facility's reactor core loading pattern operating history is bounded by the assumptions in the report as part of the license renewal application (i.e., should be a license renewal applicant action item).

RAI-7

In Section 2.4 of TR MRP-227, Rev. 0, the MRP assumes that the I&E methodology of the report is applicable to a plant that has been granted one operating license extension (i.e., through 60 years of licensed operation). The NRC staff's license renewal rule in 10 CFR Part 54 does not preclude a licensee from applying for license renewal more than once, where the RVI AMP would have been incorporated into the facility's current licensing basis prior to the second license renewal application. Thus, this assumption precludes a PWR licensee from implementing the I&E methodology in TR MRP-227, Rev. 0, beyond the expiration date associated with a PWR's first period of extended operation.

Clarify whether or not the methodology in TR MRP-227, Rev. 0, can be applied to a PWR that is applying for more than one period of extended operation and provide a basis for your conclusion. In particular, consider whether the failure modes, effects, and criticality analyses and FA methodologies incorporated within TR MRP-227, Rev. 0, methodology would meet the NRC's six definition criteria for time limited aging analyses (TLAAs) given in 10 CFR 54.3 such that a licensee might have to address them as TLAAs when seeking a second period of extended operation.

RAI-8

In Section 2.4 of TR MRP-227, Rev. 0, the MRP assumes that the design of a PWR plant applying the TR MRP-227, Rev. 0, methodology would not include any design changes beyond those identified in either general industry guidance or recommended by the original vendors. The NRC staff is aware that many of the licensees owning PWR facilities have been granted license amendments to implement measurement uncertainty recapture (MUR) power uprates, stretch power uprates, or extended power uprates for their facilities. However, it is not evident to the NRC staff whether any design changes associated with these type of power uprates would be within the scope of the MRP's term "design changes identified in general industry guidance or recommended by original vendors."

Clarify whether design changes that will need to be implemented in order to receive NRC approval of a MUR, stretch, or extended power uprate, or that have been implemented as a result of receiving NRC approval of a power uprate, are within the scope this type of assumption.

RAI-9

The middle of page 4-1 of Section 4 of TR MRP-227, Rev. 0, provides a list of program element activities for the I&E methodology. SRP-LR BTP RLSB-1 Section A.1.2.3, *Program Elements,* provides the recommended program element criteria for AMPs.

For each bulleted program criterion that is provided on page 4-1 of TR MRP-227, Rev. 0, the NRC staff requests that a reference or link be provided that matches the programmatic criterion in the bulleted list to its corresponding program element subsection in SRP-LR BTP RLSB-1 Section A.1.2.3. In addition, the NRC staff requests the criteria in the bulleted list be amended to include an aging management criterion that corresponds to the recommended program element criteria for condition monitoring programs in SRP-LR BTP RLSB-1, Section A.1.2.3.3, "*Parameters Monitored/Inspected.*"

RAI-10

Section 4.1.3, *Aging Management Methodology Qualification*, of TR MRP-227,Rev. 0, (page 4-2) provides the inspection method qualification discussion and basis for the report. The section implies that methodologies may be qualified in accordance with appropriate qualification requirements, standards, or procedures (e.g., the qualification requirements in the ASME Code), or may be qualified by only the development a technical justification to explain the applicability of the selected methodology.

Please amend the stated section to be more specific on the qualification methods that would be used to qualify a given inspection technique for implementation for both the case where an RVI component is scoped in for license renewal under one of the safety related intended functions mentioned in either 10 CFR 54.4(a)(1)(i),(ii), or (iii), and for the case where a non-safety related RVI component is scoped in for license renewal under the scoping requirement of 10 CFR 54.4(a)(2).

RAI-11

Section 4.2.5 of TR MRP-227, Rev. 0, discusses the criteria when particular aging effect indications need to be coupled to physical measurement methods. However, the discussion in Section 4.2.5 does not prescribe the physical measurment methods or techniques that would be used to quantify these aging effect indications.

Please amend Section 4.2.5 to specify the physical measurement techniques that will be used to quantify the aging effect indications for which they are credited.

RAI-12

Section 5.2 indicates that acceptance criteria for physical measurement techniques for W-designed RVI components are not included in TR MRP-227, Rev. 0, because the tolerances are available on a plant-specific or design-specific basis.

Clarify that the intent of this statement is that licensees of W-designed facilities must obtain acceptance criteria for specified physical measurements based on information in their plant's CLB. Hence, the physical measurements taken during as part of the plant's license renewal AMP must demonstrate that the condition of the affected component remains consistent with the plant's current licensing basis. Identify that this would be an applicant action item for W-designed facilities who plan to implement TR MRP-227, Rev. 0.

RAI-13

Clarify whether the acceptance criterion for eddy current (ET) inspections is based on a "pass – no pass" acceptance criterion (i.e., any ET signals indicating a relevant ET indication would fail the acceptance criterion).

RAI-14

Bolts in the some RVI components may be subject to stress relaxation resulting in reduction in preload due to thermal and irradiation effects and, as such, are inspected at every 10 year interval under the ASME Code, Section XI, ISI program. During this interval, reduction in preload in these bolts should not be large enough to cause loss of component functionality prior to the next examination. The evaluation of the need to maintain bolt preload should also consider the impact of loss of preload on vibrational fatigue damage to the bolt and/or the component itself.

Explain (based on the minimum number (percentage) of bolts/springs in each component that are required to maintain preload) how the proposed 10 year frequency is adequate to maintain functionality of each component under all design basis conditions.

Aging effects that are pertinent to some of the RVI components which are considered part of RVI AMP are addressed in GALL Tables IV B2 (Westinghouse), IV.B3 (Combustion Engineering) and IV.B4 (Babcock & Wilcox). Tables 4-1 through 4-6 in TR MRP-227, Rev. 0, are not consistent with the aforementioned GALL requirements because some aging effects of the RVI components that are addressed in GALL tables are not addressed in these tables. The following table lists the RVI components and the associated aging effects (identified in GALL requirement) which are not identified in Tables 4-1 through 4-6 of the AMP in TR MRP-227, Rev. 0. The NRC staff requests that EPRI provide the technical bases for not including these aging effects for the RVI components listed in Tables 4-1 through 4-6 of TR MRP-227, Rev. 0.

Table 1

Aging Effect	RVI Components in **Babcock & Wilcox (B&W) units** that were not listed in Tables 4-1 and 4-4 of TR MRP-227, Rev 0	GALL Report-Table ID number
Void swelling, SCC, irradiation embrittlement, irradiation-assisted stress corrosion cracking (IASCC) and, thermal and neutron embrittlement (cast austenitic stainless steel)	Control rod guide tube (CRGT) assembly –all components except Guide plates cards) and, lower flange weld	IV.B4-3 to B4-5
Loss of preload	CRGT flange to upper grid screws	IV.B4-6
Void swelling, irradiation embrittlement, and IASCC	Core barrel to Lower internal assembly bolts and top and bottom flange of core barrel cylinder	IV.B4-11, 12, 13, and 14
Void swelling, SCC, irradiation embrittlement, IASCC and loss of preload	Flow distributor assembly - All components listed in GALL with the exception of flow distributor bolts	IV.B4-23, 24, 25 and 26
Void swelling, SCC, irradiation embrittlement, IASCC, thermal embrittlement and loss of preload	Lower grid assembly--All components listed in GALL with the exception of dowel-to-guide block welds, incore guide tube spider castings and lower grid flow distributor plate, orifice plugs and lower grid and shell forgings	IV.B4-27, 28, 29,30, 31,32 and 33
Void swelling, SCC and IASCC	Plenum cover and plenum cylinder--All components listed in GALL with the exception of plenum cover weldment to rib pads and plenum cover support flange and top flange	IV.B4-34,35 and 36
Void swelling, SCC, irradiation embrittlement, and IASCC	Thermal shield	IV.B4-39, 40 and 41

Aging Effect	RVI Components in **Babcock & Wilcox (B&W) units** that were not listed in Tables 4-1 and 4-4 of the MRP-227 report	GALL Report-Table ID number
Loss of materials; pitting and crevice corrosion	RVI components	IV.B4-38
Cumulative fatigue damage—TLAA evaluation may be required	RVI components	IV.B4-37
Void swelling, SCC, irradiation embrittlement, IASCC and wear	Upper Grid Assembly-- All components listed in GALL with the exception of Alloy X-750 dowel-to-upper fuel assembly	IV.B4-42,43, 44, 45 and 46

Table 2

Aging Effect	RVI Components in **Combustion Engineering (CE)** units that were not listed in Tables 4-2 and 4-5 of the MRP-227 report	GALL Report-Table ID number
Thermal/neutron embrittlement	Cast austenitic stainless steel (CASS) in control element assembly (CEA)	IV.B3-1
Void swelling, SCC, irradiation embrittlement, IASCC and wear	CEA shroud extension shaft guides	IV.B3-3
SCC, irradiation embrittlement and loss of preload	CEA shroud bolts in bolted shroud assembly and tie rods	IV.B3-4, 5,6,7, 8,9,10,11,12 and 13
IASCC, irradiation embrittlement, wear and void swelling	Lower internal assembly--- All components listed in GALL with the exception of core support plate and fuel alignment plate(1)	IV.B3-19,20,21 and 22
Thermal/neutron embrittlement	Lower internal assembly—CASS-core support column	1V.B3-18
IASCC, irradiation embrittlement, wear and void swelling	Upper internals--All components listed in GALL	IV.B3-26,27 and 28
Loss of materials; pitting and crevice corrosion	RVI components	IV.B3-25
Aging Effect	RVI Components in CE units that were not listed in Tables 4-2 and 4-5 of the MRP-227 report	GALL Report-Table ID number
Cumulative fatigue damage—TLAA evaluation may be required	RVI components	IV.B4-24

Note (1) These aging effects are not included for this line item.

Table 3

Aging Effect	RVI Components in **Westinghouse** units that were not listed in Tables 4-3 and 4-6 of the MRP-227 report	GALL Report-Table ID number
Loss of preload	Baffle/former bolts [1]	IV.B2-5
IASCC, irradiation embrittlement, and void swelling	Core barrel flange weld[1] and core barrel outlet nozzle	IV.B2-7,8 and 9
IASCC, irradiation embrittlement, void swelling and loss of preload and wear	Lower internal assembly— All components listed in GALL with the exception of lower support bolts, lower core plate bolts and columns bodies including CASS materials	IV.B2-14,15,16,through 26
IASCC, SCC and void swelling	Control Rod Guide Tube Assembly CRGT)[2] Guide tube bolts, guide tube support pins	IV.B2-27-29[3][4] and Note
IASCC, irradiation embrittlement, void swelling and thermal embrittlement (CASS only)	Upper support column	IV.B2-35-37
Loss of preload	Upper support column bolts	IV.B2-38
IASCC,SCC and void swelling	Upper support column bolts, alignment pins and fuel alignment pins	IV.B2-39 and 40
IASCC,SCC and void swelling	Upper support plate, upper core plate hold down spring	IV.B2-41 and 42
Loss of materials; pitting and crevice corrosion	RVI components	IV.B4-32
Cumulative fatigue damage—TLAA evaluation may be required	RVI components	IV.B4-31

Note 1-- This aging effect is not included for this line item.

Note 2—In GALL this is referenced as rod control cluster assembly.

Note 3---AMP is required as a license renewal action item, for guide tube support pins—Reference—Section 4.1 item 4 of the NRC staff's safety evaluation for WCAP -14577, Revision 1, "License Renewal Evaluation: Management for Reactor Vessel Internals," report.

Note 4—Explanation is requested for not including austenitic stainless steel material used in guide tube support pins.

ELECTRIC POWER RESEARCH INSTITUTE

MRP Materials Reliability Program MRP 2010-041

(via email)

July 12, 2010

Document Control Desk
U.S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852

SUBJECT: EPRI MRP Final Responses to: REQUEST FOR ADDITIONAL INFORMATION RE: ELECTRIC POWER RESEARCH INSTITUTE TOPICAL REPORT 1016596,'MATERIALS RELIABILITY PROGRAM: PRESSURIZED WATER REACTOR INTERNALS INSPECTION AND EVALUATION GUIDELINES (MRP-227-REV. 0)' (TAC NO. ME0680), November 12, 2009

To Whom It May Concern:

Enclosed are two copies of the subject document. This document was originally provided to the Document Control Desk in draft form via EPRI Materials Reliability Program Letter MRP 2010-031 dated April 20, 2010. There are no changes to the responses in the enclosure to this letter from the draft material provided previously.

If you have any questions on this item, please contact Christine King (cking@epri.com, 650-855-2164), or Chuck Welty (cwelty@epri.com, 650-855-2371).

Sincerely,

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Final
Responses to 3rd Set RAIs
On MRP-227, Rev 0

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MRP-227 RAI Set #3 Responses
07/08/10

Titles of MRP Reports Referenced in MRP-227 or Referred to in RAI Responses

MRP #	Title	EPPRI #
MRP-128	*Materials Reliability Program: Characterization of Decommissioned PWR Vessel Internals Material Samples – Material Certification, Fluence, and Temperature,* 2003	1008202
MRP-134	*Materials Reliability Program: Framework and Strategies for Managing Aging Effects in Reactor Internals,* 2005	1008203
MRP-135 - Rev. 1	*Materials Reliability Program: Development of Material Constitutive Model for Irradiated Austenitic Stainless Steel,* 2009	1018291
MRP-156	*Materials Reliability Program: Pressurized Water Reactor Issue Management Table, PWR-IMT, Consequence of Failure,* 2005	1012110
MRP-157	*Materials Reliability Program: Updated B&W Design Information for the Issue Management Tables,* 2005	1012132
MRP-175	*Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values,* 2005	1012081
MRP-189 - Rev. 1	*Materials Reliability Program: Screening, Categorization, and Ranking of B&W-Designed PWR Internals,* 2009	1018292
MRP-190	*Materials Reliability Program: Failure Modes, Effects, and Criticality Analysis of B&W-Designed PWR Internals,* 2006	1013233
MRP-191	*Materials Reliability Program: Screening, Categorization and Ranking of Reactor Internals of Westinghouse and Combustion Engineering PWR Designs,* 2006	1013234
MRP-210	*Materials Reliability Program: Fracture Toughness Evaluation of Highly Irradiated PWR Stainless Steel Internal Components,* 2007	1016106
MRP-211	*Materials Reliability Program: PWR Internals Age-Related Material Properties, Degradation Mechanisms, Models, and Basis Data – State of Knowledge,* 2007	1015013
MRP-228	*Materials Reliability Program: Inspection Standard for Reactor Internals,* 2008	1016609

MRP-229 - Rev. 1	*Materials Reliability Program: Functionality Analysis for B&W-Designed Representative PWR Internals,* 2009	1019090
MRP-230 - Rev. 1	*Materials Reliability Program: Functionality Analysis for Westinghouse & CE-Designed Representative PWR Internals,* 2009	1019091
MRP-231-Rev. 1	*Materials Reliability Program: Aging Management Strategies for B&W-Designed PWR Internals,* 2009	1019092
MRP-232	*Materials Reliability Program: Aging Management Strategies for Westinghouse and Combustion Engineering PWR Internals,* 2008	1016593

RAI 3-1 It is not evident to the NRC staff whether or not the inspection and evaluation (I&E) methodology in TR MRP-227-Rev. 0, or in other MRP report methodologies that support the TR MRP-227, Rev. 0, methodology, are:

> a) in compliance with the scoping, screening, and aging management requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 54, or;
> b) in conformance with the NRC staff's aging management recommendations in NUREG-1800, Revision 1, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," (SRP-LR), Section A.1, "Aging Management Review – Generic (Branch Technical Position (BTP) RLSB-1)."[1]

Perform a comparison of the recommended I&E methodology in TR MRP-227, Rev. 0, and its supporting MRP aging management methodologies to the NRC staff's recommended aging effect identification and management guidance in SRP-LR BTP RLSB-1. Demonstrate the compliance of the TR MRP-227, Revision 0, methodology with the requirements of 10 CFR Part 54 and with the NRC staff's recommendations in SRP-LR BTP RLSB-1. In addition, justify any non-compliances of your proposed methodology with the stated requirements in 10 CFR Part 54 or non-conformances with the recommendations in SRP-LR BTP RLSB-1.

Response: MRP-227, Revision 0 is intended to provide the technical basis for revisions to the NRC Standard Review Plan for License Renewal (NUREG-1800, Revision1). In particular, MRP-227, Revision 0 provides information required to augment and complete the generic reactor internals Aging Management Review (AMR) and Aging Management Plan outlined in the Generic Aging Lessons Learned (GALL) Report (NUREG-1801, Revision 1). NUREG-1800, Revision 1, and NUREG-1801, Revision 1 outline an acceptable process for implementing the aging management requirements of Title 10 of the *Code of Federal Regulations* Part 54 (10 CFR Part 54). In addition, the processes used in developing MRP-227, Revision 0 are in conformance with the recommendations in SRP-LR BTP RLSB-1.

It is important to note that NUREG-1801, Revision 1 specifically states that:

> *The GALL Report does not address scoping of structures and components for license renewal. Scoping is plant specific, and the results depend on the plant design and current licensing basis. The inclusion of a certain structure or component in the GALL Report does not mean that this particular structure or component is within the scope of license renewal for all plants. Conversely, the omission of a certain structure or component in the GALL Report does not mean that this particular structure or component is not within the scope of license renewal for any plants.*

The reactor internals components addressed in MRP-227 are defined in Section 2.3, which states that:

> *These guidelines are applicable to the reactor internal structural components; they do not address fuel assemblies, reactivity control assemblies, or welded attachments to the reactor vessel.*

[1] Henceforth, the reference to this Branch Position in this set of RAIs will be abbreviated as SRP-LR BTP RLSB-1.

In other words, no items are screened out of MRP-227, Revision 0, other than fuel assemblies, reactivity control assemblies, and welded attachments to the reactor vessel.

The scoping and screening requirements of 10 CFR Part 54 are specifically discussed in Section 2.1 of NUREG-1800, Revision 1. Section 3.1 of the same document recognizes the reactor internals AMR outlined in the current version of the GALL Report. This generic AMR was developed on the assumption the reactor internals assembly as a whole is considered to be a passive, safety related structure. Therefore, it is inferred that all reactor internals components are included under the scope of 10 CFR 54 and should be included within the AMR scope. Similar statements are included in aging management evaluations of reactor internals, such as WCAP-14577 Rev.1-A and BAW-2248A, which have been accepted as generic references for license renewal programs by the NRC. MRP-227, Revision 0 is consistent with this understanding. Complete lists of components included in the generic evaluations for B&W, CE and Westinghouse plants are provided in MRP-189, Revision 1 and MRP-191. However, MRP-227 does not address the scoping and screening requirements of 10CFR54 on an individual plant basis because such actions are part of a licensee's License Renewal Application.

Section 3.01 of the SRP-LR (NUREG-1800, Revision 1) defines the AMR process as follows:

> *This AMR consists of identifying the material, environment, aging effects, and the AMP(s) credited for managing the aging effects.*

As part of the effort to support the development of the MRP-227, Revision 0 guidelines, a comprehensive evaluation of material, environment and aging effects for PWR reactor internals components was completed, as represented by MRP-189, Revision 1 and MRP-191. This effort was similar to that included in previous generic AMRs. The results of those evaluations, which are reflected in the inspection recommendations outlined in the disposition tables provided in Section 3 of MRP-227, Revision 0 (Tables 3-1, 3-2 and 3-3) are consistent with the previous studies and the current GALL tables.

An AMP is required when an effect is identified that requires specific management activities. For components within the scope of the regulation, 10 CFR 54 Section 21 requires an Integrated Plant Assessment (IPA) to assure:

> *(3) For each structure and component identified in paragraph (a)(1) of this section, demonstrate that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation.*

The MRP-227, Revision 0 IPA methodology is based on a detailed evaluation to characterize the components within the structure subject to aging effects. This process implements the guidance given in SRP-LR BTP RLSB-1, Section A.1.2.1:

> *The determination of applicable aging effects is based on degradations that have occurred and those that potentially could cause structure and component degradation. The materials, environment, stresses, service conditions, operating experience, and other relevant information should be considered in identifying applicable aging effects. The effects of aging on the intended function(s) of structures and components should also be considered.*

Section 3.1.2.2 of the SRP-LR (AMR Results for Which Further Evaluation is Recommended by the GALL Report) contains five subsections with the following statement:

> *The GALL Report recommends no further aging management review if the applicant provides a commitment in the FSAR Supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.*

Those five subsections are:

3.1.2.2.6 *Loss of Fracture Toughness due to Neutron Irradiation Embrittlement and Void Swelling*

3.1.2.2.9 *Loss of Preload due to Stress Relaxation*

3.1.2.2.12 *Cracking due to Stress Corrosion Cracking and Irradiation-Assisted Stress Corrosion Cracking (IASCC)*

3.1.2.2.15 *Changes in Dimensions due to Void Swelling*

3.1.2.2.17 *Cracking due to Stress Corrosion Cracking, Primary Water Stress Corrosion Cracking, and Irradiation-Assisted Stress Corrosion Cracking*

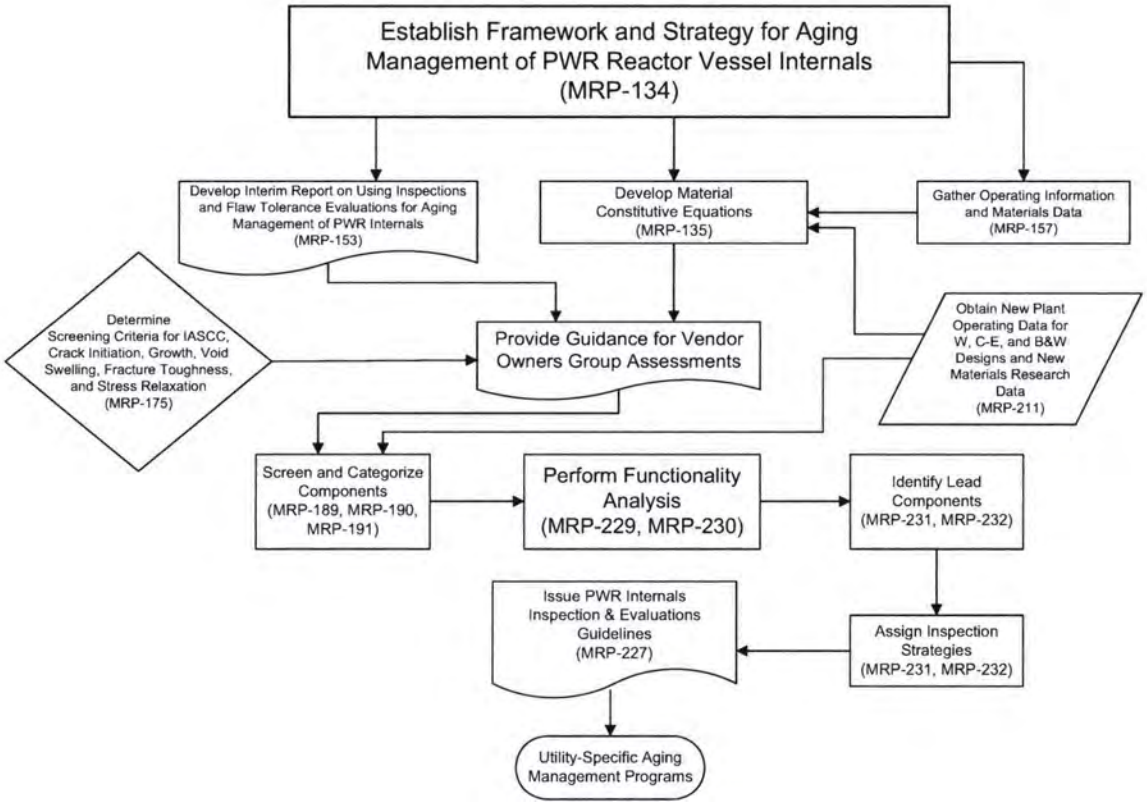
MRP-227, Revision 0 summarizes the results of industry programs for investigating and managing these aging effects in reactor internals and provides recommendations for the implementation of those results. A graphical representation of the process is depicted in MRP-227, Revision 0 Figure 2-1. The MRP-227, Revision 0 methodology begins with an extensive effort to identify materials, environment, stresses, service conditions, operating experience and other relevant information as required by SRP-LR BTP RLSB-1. This initial effort is described in MRP-134, MRP-135, MRP-153, MRP-157 and MRP-211. The second step in the process defines screening criteria that can be used to identify conditions that can potentially result in component degradation. Screening criteria were developed for eight relevant degradation mechanisms. These relevant degradation modes included the mechanisms identified in the GALL as "AMR Results for Which Further Evaluation is Recommended." Each component was compared to the MRP-175 screening criteria to identify applicable aging effects. The results of the screening process are reported in MRP-189, Revision 1 and MRP-191. This screening process is integral to the AMR and should not be confused with the screening process used to identify passive components per 10 CFR Part 54 21(a)(1).

Also in accordance with SRP-LR BTP RLSB-1, the effects of aging on the intended function(s) of the structure and components was evaluated using a FMECA (Failure Modes, Effects and Criticality Analysis) process. This FMECA process adds a level of further review that is extremely useful in prioritizing components and component

groupings prior to aging management program definition. SRP-LB BTP RLSB-1 specifically states that: "The risk significance of a structure or component could be considered in evaluating the robustness of an aging management program."

The transition from "applicable aging management effects" represented by MRP-189, Revision 1, MRP-190, and MRP-191 to "aging management program elements" is provided by MRP-231 and MRP-232, from which the requirements of MRP-227, Revision 0 were extracted and refined. The inspection guidelines provided in Section 4 of MRP-227, Revision 0 provide the basis for the program elements required in an AMP. Appendix A of MRP-227, Revision 0 summarizes the AMP Program Attributes required by NUREG-1800, Revision 1. The MRP-227, Revision 0 document is intended to provide the technical basis for an AMP. This appendix describes the extent to which the MRP-227 guidelines and supporting documents satisfy the AMP requirements. The MRP has agreed to work with the NRC to have these AMP recommendations incorporated in the upcoming GALL revision.

Figure 2.1 from MRP-227



RAI 3-2 TR MRP-227, Rev. 0, does not make reference to Nuclear Energy Institute (NEI) Report No. NEI 95-10, Revision 6,[2] 2 as an applicable industry license renewal report; nor does TR MRP-227, Rev. 0, provide any discussion on how it will be used to conform to the recommendations in NEI 95-10, Rev. 6, when implementation of the I&E recommendations are applied as license renewal activities and methods for aging management of RVI components Discuss how and provide the basis for why the recommended I&E scoping/screening process, inspection activities, evaluation methods and acceptance criteria in TR MRP-227, Rev. 0, are considered to be in conformance with the recommendations in NEI 95-10, Rev. 6, paying particular attention to conformance with Chapters 2, 3, and 4 of the NEI 95-10, Rev. 6, and the license renewal processes identified therein for implementation in Figures 2.0-1, 3.0-1, 4.1-1, 4.2-1, and 4.3-2.

Response: NEI 95-10, Revision 6, provides useful information and guidance to license renewal applicants on the steps to be followed in order to meet 10 CFR Part 54 requirements. The document was used by the team that developed MRP-227 and the supporting documentation. The basic process is outlined in the identified implementation figures.

The overall license renewal process is outlined in Figure 2.0-1. There are two major branches in Figure 2.0-1. The first branch describes aging management activities including the development of AMRs and associated AMPs. The second branch describes the process used for Time Limited Aging Analysis (TLAA). MRP-227, Revision 0 and its supporting documents provide the technical basis for aging management of systems structures and components within the reactor internals. This objective of the MRP effort was to provide documents that would support utilities in the implementation process and allow appropriate revisions to the NUREG-1801, Revision 1 (GALL) report. While MRP-227, Revision 0 is not intended to follow the implementation process precisely, it is certainly supports the aging management branch of the license renewal process. Its recommendations are consistent with the intent of the process and assure utility conformance in the license renewal process.

The method used to identify systems, structures and components within the scope of the license renewal rule is described in Section 3 of NEI 95-10 and outlined in Figure 3.0-1. Note that the process described in Figure 3.0-1 does not distinguish between systems, structures and components. The footnote to the rule in the Federal Register contains the following clarification:

> *The Commission intends that the phrase, ''systems, structures, and components'' applies to the matters involving the discussions of the overall renewal review, the specific license renewal scope (§ 54.4), time-limited aging analyses (§ 54.21(c)), and the license renewal finding (§ 54.29). The phrase, ''structures and components'' applies to matters involving the integrated plant assessment (IPA) required by § 54.21(a) because the aging management review required within the IPA should be a component and structure level review rather than a more general system level review.*

[2] NEI 95-10, Revision 6, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54-The License Renewal Rule." NEI 95-10, Rev. 6, has been endorsed for use in Regulatory Guide 1.188, "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses"[Sept. 2005].

The intention of this section is clearly to provide flexibility in the determination of systems, structures and components considered in the scope of the rule. As indicated in the response to RAI 3-1, the scope statement of MRP-227 includes the entire reactor internals structure. The function of the reactor internals structure is defined in MRP Section 3.1:

The functions of PWR internals are to:

1. *provide support, guidance, and protection for the reactor core;*
2. *provide a passageway for the distribution of the reactor coolant flow to the reactor core;*
3. *provide a passageway for support, guidance, and protection for control elements and in vessel/core instrumentation; and*
4. *provide gamma and neutron shielding for the reactor vessel.*

As these functions are critical to the operation of the reactor, the reactor internals assembly as a whole is considered safety related. Therefore, the reactor internals are within the scope of 10 CFR 54.4(a). The functions of the system have been identified per the requirements of 10 CFR 54.4(b). More detailed considerations of component function come under Section 4 of NEI 95-10.

Figure 4.1-1 describes the NEI 95-10 process for identifying specific structures and components and their function. Both the SRP-LR (NUREG-1800, Revision 1) and the GALL Report (NUREG-1801, Revision 1) identify the reactor internals as part of the Reactor Vessel, Internals and Reactor Coolant system. The reactor internals include several recognized structures (e.g., core barrel assembly, control rod guide tubes). However, the MRP analysis was completed on a component by component basis. The functions of the components and their relevance to the function of the reactor internals as a whole are described in MRP-189, Revision 1 and MRP-191. Table 2.1-5 of NUREG-1800, Revision 1 identifies the reactor internals as a "Structure, Component or Commodity that meets the Requirements of 10CFR54.21(a)(1)(i)". The reactor internals components are all passive and are not replaced on a regular basis and are, therefore, by the criteria of 10CFR54.21(a)(1)(i) and (ii), subject to management review. The NEI 95-10 process suggests that commodity groupings may be applied to the items requiring management review. The structure of the MRP review does contain several examples of large groupings of components. For instance the baffle bolting is listed as a single component item. In practice there may be hundreds of baffle bolts and there may be large variations in the stresses and irradiation exposure of the individual bolts. Similar variations may exist for reactor internals structures such as the lower support columns or the control rod guide tube assemblies. The structure of the analysis should be evident in MRP-189, Revision 1 and MRP-191.

A large portion of the MRP effort has been devoted to identification of those aging effects that require management. The MRP process was consistent with the upper portion of Figure 4.2-1 from NEI 95-10. The screening and FMECA processes described in the response to RAI 3-1 are consistent with the NEI 95-10 process. Tables 3.1, 3.2 and 3.3 of MRP-227 identify the relevant aging effects for the reactor internals.

The process used by the utility to evaluate available AMPs is described n Figure 4.3-2. The intent of the MRP effort was to define AMP options for the anticipated revision of NUREG-1801, Revision 1. Section 4.3.2 outlines the general requirements for a Plant

Specific Aging Management Program. The inspection recommendations outlined in MRP-227, Revision 0 meet these general requirements. The bases for the inspection strategies are outlined in MRP-231 and MRP-232.

Figure 2.0-1
LICENSE RENEWAL IMPLEMENTATION PROCESS