ELECTRIC POWER
RESEARCH INSTITUTE

Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)

2011 TECHNICAL REPORT

Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)

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Final Report, December 2011

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THE FOLLOWING ORGANIZATION, UNDER CONTRACT TO EPRI, PREPARED THIS REPORT:

MRP Reactor Internals Inspection and Evaluation Guidelines Core Group

NOTE

For further information about EPRI, call the EPRI Customer Assistance Center at 800.313.3774 or e-mail askepri@epri.com.

NRC SAFETY EVALUATION

In accordance with an NRC request, the NRC Safety Evaluation immediately follows this page. Other NRC and EPRI Material Reliability Program correspondence on this subject are included in the appendices.

Note: The changes proposed by the NRC in the Safety Evaluation as well those proposed by the EPRI Materials Reliability Program in response to NRC Requests for Information (RAIs) have been incorporated into the current version of the report (MRP-227-A).

December 16, 2011

Neil Wilmshurst
Vice President and Chief Nuclear Officer
Electric Power Research Institute
1300 West W. T. Harris Boulevard
Charlotte, North Carolina 28262-8550

SUBJECT: REVISION 1 TO THE FINAL SAFETY EVALUATION OF ELECTRIC POWER RESEARCH INSTITUTE (EPRI) REPORT, MATERIALS RELIABILITY PROGARM (MRP) REPORT 1016596 (MRP-227), REVISION 0, "PRESSURIZED WATER REACTOR (PWR) INTERNALS INSPECTION AND EVALUATION GUIDELINES" (TAC NO. ME0680)

Dear Mr. Wilmshurst:

By letter dated January 12, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML090160204), the EPRI submitted for U.S. Nuclear Regulatory Commission (NRC) staff review and approval MRP Report 1016596 (MRP-227), Revision 0, "Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines." On June 22, 2011, the NRC issued Revision 0 of the Safety Evaluation (SE) for MRP-227 (ADAMS Accession No. ML111600498). This letter transmits Revision 1 of the SE.

Topical Report (TR) MRP-227, Revision 0, contains an updated discussion of the technical basis for the development of an aging management program (AMP) for reactor vessel internal components in the PWR vessels supplied by Westinghouse, Babcock and Wilcox and Combustion Engineering. TR MRP-227, Revision 0, provides inspection and evaluation guidelines as part of an AMP for use by the licensees. Revision 1 of the SE incorporates technical changes required to ensure the final TR (-A version) includes all NRC required changes.

The NRC staff has found that TR MRP-227 is acceptable for referencing in licensing applications for PWR internals inspection and evaluation to the extent specified in the enclosed final Revision 1 of the SE. The final Revision 1 SE defines the basis for acceptance of the TR. The staff's final evaluation of the MRP-227, Revision 0 report, including eight plant-specific action items and seven conditions is enclosed.

Our acceptance applies only to material provided in the subject TR. We do not intend to repeat our review of the acceptable material described in the TR. When the TR appears as a reference in license amendment requests or license renewal (LR) applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests or references to this TR in LR applications that deviate from this TR will be subject to a plant-specific review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC public website, we request that EPRI publish an accepted version of this TR within three months of receipt of this letter. The accepted version shall incorporate; the changes outlined in the SE, and this letter and the enclosed final SE after the title page. Also, it must contain historical review information, including NRC requests for additional information and your responses. The accepted version shall include a "-A" (designating accepted) following the TR identification symbol. If future changes to the NRC's regulatory requirements affect the acceptability of this TR, EPRI and/or licensees referencing it will be expected to revise the TR appropriately, or justify its continued applicability for subsequent referencing.

Sincerely,

/RA/

Robert A. Nelson, Deputy Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 669

Enclosure:
Final SE

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enclosed final SE after the title page. Also, it must contain historical review information, including NRC requests for additional information and your responses. The accepted version shall include a "-A" (designating accepted) following the TR identification symbol. If future changes to the NRC's regulatory requirements affect the acceptability of this TR, EPRI and/or licensees referencing it will be expected to revise the TR appropriately, or justify its continued applicability for subsequent referencing.

Sincerely,

/RA/

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Enclosure:
Final SE

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REVISION 1 TO THE SAFETY EVALUATION BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

MATERIALS RELIABILTY PROGRAM: PRESSURIZED WATER REACTOR INTERNALS

INSPECTION AND EVALUATION GUIDELINES (MRP-227, REVISION 0)

PROJECT NO. 669

1.0 INTRODUCTION

The objective of the topical report (TR) process is, in part, to add value by improving the efficiency of other licensing processes, for example, the process for reviewing license amendment requests from commercial operating reactor licensees. The purpose of the U.S. Nuclear Regulatory Commission (NRC) TR program is to minimize industry and NRC time and effort by providing for a streamlined review and approval of a safety-related subject with subsequent referencing in licensing actions, rather than repeated reviews of the same subject.

A TR is a stand-alone report containing technical information about a nuclear power plant safety topic, which meets the criteria of a TR. A TR improves the efficiency of the licensing process by allowing the NRC staff to review a proposed methodology, design, operational requirements, or other safety-related subjects that will be used by multiple licensees, following approval, by referencing the approved TR. The TR provides the technical basis for a licensing action.

During the review of the Electric Power Research Institute's (EPRI) TR MRP-227, Revision 0, the NRC staff found that, in general, the TR meets the objectives of a TR and reinforces previously established NRC regulations and guidelines as noted within this safety evaluation (SE). The NRC has evaluated this TR against the criteria of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, and has determined that it does not represent a backfit. Specifically, NRC staff technical positions outlined in this SE are consistent with the aforementioned regulations and established staff positions, while providing more detailed discussion concerning the methodology and data required supporting reactor internals inspections. This SE endorses staff positions previously established through licensing actions and interactions with industry.

1.1 Background

By letter dated January 12, 2009 (Reference 15), the EPRI submitted for NRC staff review and approval Materials Reliability Program (MRP) Report 1016596, Revision 0, "Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines" (MRP-227).

By letter dated March 2, 2010 (ADAMS Accession No. ML100640166), EPRI informed the NRC that MRP-227 Revision 0, was made publicly available and is no longer proprietary.

Enclosure

By letter dated June 22, 2011 (Reference 19), the NRC issued the final SE Revision 0 for TR MRP-227 Revision 0. Revision 1 of the SE incorporates technical changes required to ensure the final TR (-A version) includes all NRC required changes.

MRP-227 contains a discussion of the technical basis for the development of an aging management program (AMP) for reactor vessel internal (RVI) components in PWR vessels supplied by Westinghouse, Babcock and Wilcox (B&W) and Combustion Engineering (CE). MRP-227 provides inspection and evaluation (I&E) guidelines as part of the AMP for use by the applicants/licensees.

1.2 Purpose

The NRC staff reviewed MRP-227 to determine whether its guidance will provide reasonable assurance that the I&E of the subject RVI components will ensure that the RVI components maintain their intended functions during the period of extended operation. The review also considered compliance with license renewal (LR) requirements in 10 CFR 54.21(a)(3) in order to allow licensees or applicants the option of adopting the aging management methodology described in MRP-227 as the basis for managing age-related degradation in RVI components and incorporating, by reference, the recommended guidelines into PWR Vessel Internals AMPs (or their equivalents). This option is consistent with the recommendations in AMP, XI.M16A, "PWR Vessel Internals," of the Generic Aging Lessons Learned (GALL) Report, Revision 2 (NUREG-1801, Revision 2).

1.3 Organization of the Safety Evaluation

Section 2.0 of this SE summarizes MRP-227. Section 3.0 documents the staff's evaluation and findings pertaining to the adequacy of the MRP's AMP recommendations. In particular, Section 3.0 documents staff concerns with MRP-227 and the basis for limitations and conditions being placed on the use of MRP-227 as well as licensee/applicant action items that shall be addressed by applicants/licensees who choose to implement the NRC-approved version of MRP-227. Section 4.0 summarizes the limitations and conditions and the applicant/licensee action items. Section 5.0 provides the conclusions resulting from this SE.

1.4 Regulatory Requirements

Title 10 of the *Code of Federal Regulations* (CFR) Part 54 addresses the requirements for plant license renewal. The regulation at 10 CFR 54.21 requires that each application for LR contain an integrated plant assessment (IPA) and an evaluation of time limited aging analyses (TLAAs). The IPA shall identify and list those structures and components subject to an aging management review (AMR) and demonstrate that the effects of aging (cracking, loss of material, loss of fracture toughness, dimensional changes, loss of preload) will be adequately managed so that their intended functions will be maintained consistent with the current licensing basis (CLB) for the period of extended operation as required by 10 CFR 54.29(a). In addition, 10 CFR 54.22 requires that a LR application include any technical specification (TS) changes or additions necessary to manage the effects of aging during the period of extended operation as part of the LR application.

Structures and components subject to an AMR shall encompass those structures and components that (1) perform an intended function, as described in 10 CFR 54.4, without moving parts or without a change in configuration or properties and (2) are not subject to replacement based on a qualified life or specified time period. These structures and components are referred to as "passive" and "long-lived" structures and components, respectively. The scope of components considered for inspection under MRP-227 guidance includes core support structures (typically denoted as Examination Category B-N-3 by the American Society of Mechanical Engineers (ASME) Code, Section XI) and those RVI components that serve an intended LR safety function pursuant to criteria in 10 CFR 54.4(a)(1). The scope of the program does not include active RVI components (e.g.: vent valve discs, shafts or hinge pins), or consumable items such as fuel assemblies, reactivity control assemblies, and nuclear instrumentation because these components are not typically within the scope of the components that are required to be subject to an AMP, as defined by the criteria set in 10 CFR 54.21(a)(1).

Some owners of PWR units were granted renewed licenses and some of these licensees made a commitment to conform to the recommendations specified in NUREG-1801, GALL, Revision 1, AMP XI.M16, "PWR Vessel Internals." AMP XI.M16 requires that the applicant provide a commitment in the Final Safety Analysis Review (FSAR) supplement to (a) participate in the industry programs for investigating and managing aging effects on RVI components; (b) evaluate and implement the results of the industry programs as applicable to the RVI components; and (c) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for RVI components to the NRC for review and approval. Each applicant/licensee that made a commitment to conform to the recommendation specified in NUREG-1801, Revision 1, AMP XI.M16 also made a commitment in its FSAR that it will implement the industry developed AMP for its RVI components.

If a LR applicant confirms that it will implement MRP-227 guidelines, as modified by this SE Revision 1, at its plant, then no further review of the AMP for the PWR RVI components is necessary, except as specifically identified in Section 4.0 of this SE. With these exceptions, an applicant may rely on MRP-227 for the demonstration required by Section 54.21(a)(3) with respect to the RVI components and structures within the scope of MRP-227. Under such circumstances, the staff intends to rely on the evaluation in this SE to make the findings required by 10 CFR 54.29 with respect to a particular application.

2.0 SUMMARY OF MRP-227

MRP-227 contains a discussion of the technical basis for implementing inspection requirements for PWR RVI components that are subject to any of the applicable degradation mechanisms (e.g., stress corrosion cracking (SCC), intergranular stress corrosion cracking, irradiation-assisted stress corrosion cracking (IASCC), wear, fatigue, thermal and/or neutron embrittlement, void swelling, and irradiation-enhanced stress relaxation) during the LR period. MRP-227 also provides a brief, high-level summary of flaw evaluation guidelines for RVI components that exhibit active degradation mechanisms, and establishes requirements for inspection of additional components if an active degradation mechanism is discovered (i.e., expansion of the scope of RVI component inspections). Extensive information was provided with respect to the effects of the applicable degradation mechanisms on various RVI components and the inspection requirements for these components. The following sections include a brief description of the information contained in MRP-227.

2.1 MRP-227, Revision 0 - Section 1

Section 1 of MRP-227 includes an overall synopsis related to aging management of the PWR RVI components by identifying the following steps in the MRP's process for developing the I&E guidelines: (1) development of screening criteria for the applicable degradation mechanisms; (2) screening of the different RVI components designed by Westinghouse, B&W, and CE based on the components' susceptibility to degradation; (3) functionality analyses and failure modes, effects, and criticality analyses (FMECAs) performed for the components which resulted in the binning of components into different inspection categories; and (4) development of the proposed I&E guidelines and flaw evaluation methodology.

Step (1) of this process was not discussed in MRP-227 but was documented in MRP-175, "Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values." MRP-227 also referenced MRP-211, "Materials Reliability Program: PWR Internals: Age Related Material Properties Degradation Mechanisms, Models and Basis Data," which addresses screening criteria for the degradation mechanisms in PWR RVI components. Screening of PWR RVI components for susceptibility to the degradation mechanisms was performed by establishing a set of screening criteria for each relevant degradation mechanism. The MRP-175 report provided technical data that was obtained from experiments to provide the basis that the MRP used to develop the screening criteria for different degradation mechanisms. The screening criteria for the degradation mechanisms considered in MRP-227 depend on various factors. For example, the screening factors for SCC depend on type of material and applied stress.

2.2 MRP-227, Revision 0 - Sections 2 and 3

In Sections 2 and 3 of MRP-227, the MRP provided an expanded discussion regarding steps (2) and (3) identified in Section 2.1 of this SE. In this SE, these steps, which lead up to the binning of components into inspection categories, may be referred to as the "categorization" phase of the MRP's process.

As background material, Section 3 of MRP-227 discussed the various design characteristics, and their functions, of the RVI components supplied by Westinghouse, CE, and B&W. This section also discussed potential aging effects that may result from the identified degradation mechanisms. These aging effects included: (1) various forms of cracking, (2) loss of material induced by wear; (3) loss of fracture toughness due to either individual or synergistic contributions from thermal aging or neutron irradiation embrittlement; (4) dimensional changes and potential loss of fracture toughness due to void swelling and irradiation growth; and (5) loss of preload due to either individual or synergistic contributions from thermal and irradiation-enhanced stress relaxation or creep.

Initial screening of RVI components for all three (B&W, CE, and Westinghouse) designs was based on a consideration of material properties (e.g., chemical composition) and operating conditions (e.g., neutron fluence exposure, temperature history, and representative stress levels) in order to determine the susceptibility of PWR RVI components to the applicable aging mechanisms. This resulted in the binning of these RVI components as either susceptible or not susceptible to each of the eight degradation mechanisms, based on the degradation screening criteria.

Next, the MRP performed a failure modes, effects and criticality analysis (FMECA) of the RVI components. The FMECA process was discussed in detail in 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 Combustion Engineering PWR Designs." The FMECA was a qualitative process that included expert elicitation by technical experts. Expert elicitation was used for developing the technical basis for categorization of various RVI components under different categories based on the combination of the likelihood of component degradation due to one or more of the eight degradation mechanisms, and the severity of safety consequences. Each component was assigned to one of three categories (for each degradation mechanism) ranging from insignificant effects (Category A) to potentially moderately significant effects (Category B) to potentially significant effects (Category C). Category C components were associated with higher risk in that they are more susceptible to aging degradation and the consequences of their failure are more severe. Category C components were also often considered the likely lead components for providing telltale signs of the associated aging degradation. Category B components, on the other hand, can still be susceptible to aging degradation but their consequences of failure are typically less than Category C components. Category A components are either (a) those which have been judged to be not susceptible to any of the eight degradation mechanisms or (b) those which have been judged to be somewhat susceptible to one or more aging degradation mechanisms but are not expected to lose functionality.

The MRP then performed a functionality assessment of the PWR internals components and items that would most be affected by the degradation mechanisms (i.e., preliminary Category B and C items from the FMECA). This assessment was based on representative plant designs using irradiated and aged material properties. The functionality analyses included finite element analyses on selected RVI components that were deemed to be susceptible to irradiation-induced degradation mechanisms (e.g., IASCC, neutron embrittlement, void swelling, and irradiation-induced stress relaxation) where the effects are dependent on multiple variables and develop with time to assess the evolution of degradation. The functionality analyses were used to demonstrate that although some Category C components were susceptible to one or more degradation mechanisms, the effect of the degradation mechanisms on their performance was not significant.

It should be noted that the FMECA and functionality analyses were based on the assumption of thirty years of operation with high leakage core loading patterns followed by thirty years of low leakage core loading patterns. In the U.S. PWR fleet, low leakage core loading patterns were implemented early in the unit's operating lives. Hence, MRP considered this assumption conservative. The MRP also assumed a base load operation such that the modeled plants operate at fixed power levels and do not vary power on a calendar or load demand schedule.

Industry considered the results from the FMECA and functionality analysis along with operating experience, component accessibility, and existing inspection programs to develop the recommended inspection categories for maintaining the long-term functionality of PWR RVI components. In Section 3, the MRP, based on this assessment, developed four inspection categories:

1. Primary – RVI components that are either highly susceptible to effects of aging due to any active degradation mechanism, or components that have a degree of tolerance for a

specific degradation mechanism but for which no leading highly susceptible or accessible component exists. These components are to be periodically inspected as part of a RVI component AMP.

2. Expansion – RVI components that are moderately or highly susceptible to the effects of aging due to one or more active degradation mechanisms, but for which the functionality analyses indicated that these components have a degree of tolerance to the aging effects associated with these degradation mechanisms. These components will be inspected as part of a RVI component AMP if unacceptable degradation is identified during inspections of relevant "Primary" inspection category components.

3. Existing (Programs) – RVI components that are susceptible to the effects of aging due to one or more active degradation mechanisms, but that are managed under an existing generic or plant-specific AMP currently implemented by the PWR fleet which adequately manages the aging effect. MRP-227 consistently calls this category the "Existing" inspection category, but for clarity it will be referred to as the "Existing (Programs)" inspection category in this SE.

4. No Additional Measures – RVI components that are below the screening criteria for the applicable degradation mechanisms, or were classified under this category due to FMECA and functionality analysis findings. No further action is required by MRP-227 for managing the aging of these components.

Tables 3-1 through 3-3 in Section 3 of MRP-227 summarize the proposed inspection categories for each B&W, CE, and Westinghouse RVI component that was initially placed into Categories B and C as a result of the initial screening and FMECA analyses. These tables identify the proposed inspection categories associated with each of the individual degradation mechanisms as well as the final grouping. The final I&E guidelines were based on the summary classifications contained in these tables.

2.3 MRP-227, Revision 0 - Sections 4 and 5

In Sections 4 and 5 of MRP-227, a detailed discussion regarding: (1) the examination method to be applied for a particular component based on its final categorization (see Section 2.2 of this SE); (2) qualifications for the examinations; (3) examination frequency; (4) sampling and coverage; (5) expansion scope of examination based on the extent of observed degradation; and (6) evaluation of examination results. In this SE, the staff will refer to this information as the MRP's proposed I&E guidelines for components subject to MRP-227. Tables 4-1, 4-2, and 4-3 of MRP-227 address the identification of "Primary" inspection category components, their relevant aging effects, and the type of examination methods to be used for plants designed by B&W, CE, and Westinghouse, respectively. Similar information is provided in Tables 4-4, 4-5, and 4-6 for the "Expansion" inspection category components designed by B&W, CE, and Westinghouse, respectively. Tables 4-8 and 4-9 include similar information for some components in the "Existing (Programs)" inspection category for plants designed by CE and Westinghouse, respectively. No existing generic industry programs were considered sufficient to monitor the aging effects in RVI components designed by B&W and, hence, no Table 4-7 was included. Although categorized under the "Existing (Programs)" inspection category, CE thermal shield positioning pins, CE in-core instrumentation thimble tubes, and Westinghouse guide tube support pins (split pins) were not included in Tables 4-8 and 4-9 because the adequacy of the plant-specific existing programs to manage degradation of these components for the period of extended operation could not be verified in the development of MRP-227.

The examination methods endorsed by MRP-227 include: (1) ASME Code, Section XI, visual (VT-3 and VT-1) examinations; (2) enhanced visual (EVT-1) and VT-1 examinations; (3) surface examination [eddy current testing (ET)], (4) volumetric examination using ultrasonic techniques, and (5) physical measurements. Selection of an examination method was based on the characterization of a particular degradation mechanism. It was also based on the examination method that is capable of identifying the aging effect associated with the degradation mechanism. MRP's proposed examinations are to be implemented by well-established standard procedures and these procedures are to be qualified per industry inspection standards addressed in MRP-228, "Materials Reliability Program: Inspection Standard for Reactor Internals." Some examination methods require additional qualifications per ASME Code, Section V, "Non-Destructive Examinations."

In general, the "Primary" and "Existing (Programs)" inspection category components are to be examined once during every 10-year ISI interval. Tables 4-1, 4-2, 4-3, 4-8, and 4-9 address the frequency of examinations to be used for these components in plants designed by B&W, CE, and Westinghouse. For some components (e.g., baffle bolts), MRP-227 specifically notes that the frequency of examination may be increased based on inspection results. In general, operating experience gathered from inspections conducted in accordance with the NRC-approved version of MRP-227 will be reviewed and used to update inspection requirements.

Tables 4-1, 4-2, 4-3, 4-4, 4-5, 4-6, 4-8, and 4-9 address the requirements for the examination coverage for RVI components in plants designed by B&W, CE, and Westinghouse. In addressing the coverage to be obtained when examinations are performed, MRP-227 states that for all "Primary" and "Expansion" inspection category components, one hundred percent of accessible surfaces/volumes are required to be examined, with the exception of some components for which limited accessibility is known to exist. In this case, known limited accessibility was related to the need to disassemble the RVI components in order to achieve full accessibility to all of a set of like components for examination. Types of like components with known limited accessibility included, for example, Westinghouse guide cards in control rod guide tube (CRGT) assemblies. For these sets of components, MRP-227 required an inspection sample, ranging from 10 percent to 20 percent of each subject set of like components. For the 10 percent to 20 percent sample of each set of components to be inspected, MRP-227 required that one hundred percent of the accessible surfaces/volumes be examined.

MRP-227 addressed the examination of "Expansion" inspection category components, which is based on the extent of aging degradation observed in a related "Primary" inspection category component in Tables 4-4, 4-5, 4-6, 5-1, 5-2, and 5-3. The criteria for initiating the examination of the "Expansion" inspection category components is based on the column on the linkage between the "Primary" and "Expansion" inspection category components established in these tables. In general, a single "Primary" inspection category component that is being inspected to monitor for a particular degradation mechanism may be linked to more than one "Expansion" inspection category component. The observation of degradation in the "Primary" inspection category component could trigger the need to examine the associated "Expansion" inspection category components, depending on the licensee's evaluation of the significance of observed degradation in the "Primary" inspection category component. Certain "Expansion" inspection category RVI components were determined to be completely inaccessible for examination, including the B&W core barrel cylinder (including vertical and circumferential seam welds), former plates, external baffle-to-baffle bolts and their locking devices, core barrel-to-former bolts and their locking devices, and core support shield vent valve disc shafts or hinge pins. For these inaccessible "Expansion" category components, MRP-227 stated that, when their inspection is called for based upon the observation of a degradation mechanism in the associated "Primary" inspection category component, the applicant/licensee must evaluate the continued operability of the inaccessible "Expansion" inspection category component or, alternatively, replace the component.

With regard to the evaluation of examination results, Tables 5-1, 5-2, and 5-3 and the text of Section 5 provide: (1) relevant conditions for each specified examination method and (2) general guidance on the evaluation of relevant conditions for plants designed by B&W, CE, and Westinghouse, respectively. For example, for EVT-1 examinations, the specific relevant condition identified in MRP-227 is a detectable crack on the surface of an RVI component. The acceptance criteria then provided for the relevant conditions associated with this examination method was that only the absence of a relevant condition would require no further evaluation. An acceptable process to disposition relevant conditions may include supplemental examinations, accepting the condition until the next examination, or replacement of the component. The outcome of the evaluation of the relevant condition may also affect the implementation of the examination of associated "Expansion" inspection category components.

2.4 MRP-227, Revision 0 - Section 6

Section 6 of MRP-227 provided guidance on the application of flaw evaluation methodologies to be implemented when an examination reveals the presence of a relevant condition. Various subsections in Section 6 provided details on:

1. The loading conditions to be considered when evaluating core support structures, including deadweight loads, mechanical loads, hydraulic loads, thermal loads, and loads from operating basis and safe shutdown earthquakes.

2. The requirements and limitations (based on accumulated neutron fluence) for the application of limit load evaluation methodologies for flawed RVI components. The requirements include application of limit load procedures similar to those given in ASME Code, Section XI.

3. The application of linear elastic fracture mechanics and elastic-plastic fracture mechanics for RVI components with an accumulated neutron fluence that exceeds the limit load application threshold limit.

4. The application of existing crack growth rate values for the evaluation of SCC in stainless steel components and IASCC in irradiated stainless steel components.

5. The evaluation of flaws in bolts and bolted assemblies. This includes the assessment of the functionality of bolted assemblies that may contain one or more non-functional bolts. This evaluation is to be based on the minimum number required to maintain the functionality of the assembly until the next examination.

While this evaluation guidance is included in MRP-227, it is important to note that the industry submitted TR WCAP-17096-NP (Reference 14) for staff review and endorsement. WCAP-17096 supports the inspection and evaluation guidelines outlined in MRP-227. This WCAP report is consistent with and supplemental to the guidance contained in Section 6 of MRP-227. The guidance in the WCAP will be used to evaluate component degradation that exceeds the acceptance criteria in Section 5 when it is observed during required inspections.

2.5 MRP-227, Revision 0 - Section 7

Section 7 of MRP-227 provided a summary of the implementation requirements for the guidelines described in MRP-227. The implementation requirements are defined by the latest edition of Nuclear Energy Institute (NEI) Implementation Protocol NEI 03-08, "Guidelines for the Management of Materials Issues," which includes implementation categories used in MRP-227 including: (a) "Mandatory," which requires implementation of the guidelines at all plants; (b) "Needed," which provides an option for implementing the guidelines wherever possible or implementing alternative approaches, or (c) "Good Practice," which recommends implementation of the guidelines as an option whereby significant operational and reliability benefits can be achieved at a given plant. Failure to meet a "Needed" or a "Mandatory" requirement is a deviation from the guidelines and a written justification for deviation must be prepared and approved as described in Addendum D to NEI-03-08. A copy of the deviation is sent to the MRP so that, if needed, improvements to the guidelines can be developed. A copy of the deviation is also sent, for information, to the NRC.

Section 7 of MRP-227 specified the following with respect to the implementation of specific MRP-227 guidelines:

1. Each PWR unit shall develop and document an AMP for the PWR RVI components within thirty-six months following the issuance of MRP-227-A. This is a "Mandatory" requirement.

2. Each PWR unit shall implement Tables 4-1 through 4-9 and Tables 5-1 through 5-3 of MRP-227 for the applicable design within twenty-four months following the issuance of MRP-227-A. This is a "Needed" requirement.

3. Examination of the RVI components shall comply with the MRP-228 Revision 0, "Materials Reliability Program: Inspection Standard for PWR Internals." This is a "Needed" requirement.

4. Examination results that do not meet the examination acceptance criteria defined in Section 5 of the MRP-227 guidelines shall be recorded and entered in the plant corrective action program and dispositioned.

5. A summary report of all inspections and monitoring, evaluation, and new repairs shall be provided within one hundred and twenty days of the completion of an outage during which the RVI components were examined. The summary of the examination results shall be included in an industry report that is updated every six months. This report will monitor the industry progress on the AMP related to PWR RVI components and it will also list the emerging operating experience. This is a "Good Practice" requirement.

2.6 MRP-227, Revision 0 - Appendix A

Appendix A addresses how the AMP defined in MRP-227 meets specific AMP attributes as defined by NUREG-1801, the LR GALL report. Specifically, Appendix A discusses how the MRP-227 program meets the "Scope of Program" (Attribute 1 from NUREG-1801), "Parameters Monitored" (Attribute 3 from NUREG-1801), and "Detection of Aging Effects" (Attribute 4 from NUREG-1801). Appendix A also stated that supplementary information shall be provided by the applicants/licensees to satisfy all the NUREG-1801 AMP requirements for the remaining program elements when implementing MRP-227.

3.0 STAFF EVALUATION

The staff reviewed MRP-227 to determine whether the scope of RVI components in MRP-227 were consistent with those that would need to be subject to an aging management review, as required in accordance with the provisions in 10 CFR 54.21(a)(1). The staff determined that, consistent with the requirements in 10 CFR 54.21(a)(1), the scope of MRP-227 includes all passive, long-lived Westinghouse-design, CE-design, and B&W-design RVI components that need to be within the scope of LR (refer to the LR scoping requirements in 10 CFR 54.4). The staff also determined that, consistent with the aging management review requirements in 10 CFR 54.21(a)(1), the scope of MRP-227 does not include any components that involve movable parts or a change in configuration (i.e., active RVI components, such as B&W-design vent valve discs, shafts or hinge pins, or RVI nuclear instrumentation) or components that would be subject to replacement based on a qualified life or specified time period (i.e., consumable items, such as fuel assemblies or reactor control assemblies).

The staff also reviewed MRP-227 to determine if it demonstrated that the effects of aging on the RVI components covered by MRP-227 would be adequately managed so that the components' intended functions would be maintained consistent with the CLB for the period of extended operation, in accordance with 10 CFR 54.21(a)(3). Besides the IPA, 10 CFR Part 54 requires an evaluation of TLAAs, in accordance with 10 CFR 54.21(c). The staff reviewed MRP-227 to determine if the TLAAs covered by MRP-227 were evaluated for LR in accordance with 10 CFR 54.21(c).

During its review of MRP-227, the staff issued four sets of requests for additional information (RAIs) that addressed technical issues. The details of the staff's RAIs and the corresponding responses are available in ADAMS (proprietary version). However, the staff did not include all the RAIs and the MRP's responses in this SE; it included only those salient RAIs and MRP responses that address specific points of emphasis. References 16 through 18 contain all of the staff's technical RAIs and the MRP's responses. In addition, a draft version of the NRC's MRP-227 SE Revision 0 (ADAMS Accession No. ML110820773) was posted for public comment on April 11, 2011, for 30 days. All comments received during this public comment period were from industry, and were reviewed and considered during the development of this final SE.

3.1 Evaluation of MRP-227, Revision 0 - Section 1

The staff reviewed Section 1 of MRP-227 and accepts the approach used by the MRP to develop the screening criteria for initially binning the RVI components into Category A, B, and C. In this section, the MRP provided technical data that was used as the basis for the screening criteria for different degradation mechanisms. The screening criteria were based on: (1) type of material used in RVI components, (2) operating stress levels, and in some cases, (3) neutron fluence values. For example, IASCC screening criteria were established by (1) type of material, (2) threshold limit of neutron fluence value and (3) stress values. The threshold limits for neutron fluence and stress levels were developed by valid research data that is widely used by the industry. Similar criteria were developed for the other degradation mechanisms. The NRC staff has not officially reviewed the technical basis for the screening criteria that is contained in MRP-175 and MRP-211. Therefore, the NRC staff does not specifically endorse the screening criteria used in MRP-227. However, the MRP-227 strategy of identifying "Primary" inspection components based on the relative likelihood of degradation compared to other components diminishes the importance of the specific screening criteria values used in MRP-227.

3.2 Evaluation of MRP-227, Revision 0 - Sections 2 and 3

The staff's review of Sections 2 and 3 of MRP-227 resulted in the staff, in principle, accepting the MRP's categorization process for the development of an AMP for the RVI components. The MRP considered susceptibility of RVI components to one or more degradation mechanisms and the safety consequences as a result of the failure of the RVI components. However, the staff identified some concerns with the MRP's categorization process and/or its application. The staff's evaluation of the MRP's process is provided below, focusing on the staff's concerns which led to the imposition of conditions and limitations on the use of MRP-227 and plant-specific action items associated with the use of MRP-227 (as discussed in Section 4 of this SE).

3.2.1 General Evaluation of MRP's Categorization Process - Initial Screening, FMECA, Functionality Analyses, and the Assigning of Components to Inspection Categories

In Sections 2 and 3 of MRP-227, the MRP discussed their categorization process for various RVI components. The categorization process (i.e., initial screening, FMECA, and functionality analyses) described in MRP-227 provides an adequate approach for identifying the degradation mechanisms for RVI components within the scope of LR. Those components that were assessed to be most affected by one or more of the degradation mechanisms addressed in Section 2.0 of this SE were binned under Category C, those components that were expected to be moderately affected by the degradation mechanisms were binned under Category B, and components that were expected to be unaffected by the degradation mechanisms were binned under Category A. The initial screening process entailed evaluation of material properties, corrosion resistance of materials, the effect of neutron fluence on some components, and loading conditions. The staff concluded that the MRP had adopted a systematic approach in the initial screening of the RVI components into various categories, and the staff accepts this approach.

The staff, in principle, also agrees with the technical basis used in the development of the recommended component inspection groupings identified in Section 2.2 of this SE based, in part, on using FMECA and functionality analysis. However, in its review of the FMECA process described in MRP-190 and MRP-191 and the functionality analyses described in MRP-229 and MRP-230, the staff identified concerns with the MRP's approach. Some of the staff's concerns were resolved via MRP responses to staff RAIs, while concerns that were not adequately resolved are reflected in plant-specific action items and/or conditions and limitations on the use of MRP-227. Examples of significant staff concerns that were resolved are given in the following paragraphs, and those that were not adequately resolved are addressed in Sections 3.2.2, 3.2.3, and 3.2.4 of this SE.

The staff requested that the MRP address the impact of the potential aging effects on the RVI components and reactor system performance in transient and accident conditions. In its response, the MRP provided information to demonstrate that component loadings assumed in the FMECA process included normal operating loads and, in some cases, both normal operating loads and transient loadings. The MRP stated that the expert elicitation process also assessed the safety implications of potentially failed components, and that it could be inferred that the non-escalation of consequences was considered during the FMECA process. The MRP also stated that, as discussed in MRP-190, the expert elicitation process explicitly considered whether the aging effects considered in the FMECA process would result in more severe consequences if a design basis transient occurred. Further, the MRP indicated that if degradation is found during inspections, the subsequent evaluation of the degraded component's integrity is performed using the guidance in WCAP-17096-NP, Revision 2 which is currently under staff review. The WCAP evaluation requires that acceptable component performance be demonstrated under all design basis conditions such that the licensing basis is maintained. Component repair or replacement is required if this evaluation demonstrates that the licensing basis cannot be maintained. The staff accepts this response and this issue is resolved pending the review of WCAP-17096-NP, Revision 2.

The staff also had concerns associated with some of the FMECA results and the outcome of some of the functionality analyses. Some RVI components that were originally identified for potential aging degradation due to single or multiple aging degradation mechanisms (Categories B and C) were placed under the "No Additional Measures" inspection category as a result of the FMECA or functionality analyses. The staff was concerned that these components could be subject to damage and possible deterioration of the original mechanical properties due to aging degradation. Hence, the structural integrity of these RVI components could be challenged under licensing basis loading conditions. The MRP provided a few examples and included acceptable technical justification for categorizing some RVI components from Category B and C to the "No Additional Measures" Inspection category. The examples include: (1) Westinghouse bottom mounted instrumentation cruciforms, and (2) Westinghouse lower core plate fuel alignment bolts. The staff accepts the response and considers this issue resolved.

3.2.2 High Consequence Components in the "No Additional Measures" Inspection Category

During the review of the FMECA process, the staff identified a concern regarding the categorization of some of the RVI components whose failure could cause significant safety consequences. In some cases, the MRP placed these components under the "No Additional Measures" inspection category. The following paragraphs discuss the categorization of these high consequence RVI components. The relevant high consequence components are: (1) the upper core plate and lower support forging or casting in Westinghouse-designed reactors, and (2) the lower core support beams, core support barrel assembly (CSBA) upper cylinder and CSBA upper core barrel flange in CE-designed reactors.

CE and Westinghouse RVI components were grouped in risk categories as part of the FMECA based on the combination of (1) their likelihood of failure and (2) a qualitative assessment of the potential for core damage associated with their failure. The staff's concern is related to those components that were qualitatively assessed as having a "high" potential for core damage associated with their failure (i.e., high consequence components) that are not already identified for inspection within the "Primary" or "Expansion" categories. An RVI component was considered to have a "high" potential for core damage when it was believed that some core damage could result from failure of the component, for example, related to the inability to safely shutdown the reactor. The likelihood of degradation in these components was typically assessed in MRP-227 as being "low." A component was identified as having a "low" likelihood of failure when there were no known failures of this component based on operating experience, and it is believed that the failure is unlikely to occur during extended period of operation. A similar approach was used for the B&W components, although different terminology was used. For B&W components, those in "Risk Band III" were understood to be similar to the combination of "high" potential for core damage associated with their failure and a "low" likelihood of failure from the Westinghouse/CE characterization.

The staff determined that the MRP did not provide an adequate justification regarding how these high consequence/low likelihood of failure RVI components were assigned to the "No Additional Measures" inspection category. The staff is concerned that these components could be subject to loss of structural integrity due to one or more degradation mechanisms. To ensure that the structural integrity and functionality of these RVI components are maintained under all licensing basis conditions during the period of extended operation, the staff has determined that these components shall be included in the "Expansion" inspection category in the NRC-approved version of MRP-227. The staff recognizes that several or all of these components are subject to ASME Code, Section XI VT-3 inspections. However, the examination method to be used for additional inspections of "Expansion" inspection category components triggered by degradation in the "Primary" inspection category components to which they are linked shall be consistent with the examination method used to identify the "Primary" component degradation. The staff has identified "Primary" inspection category links for the upper core plate and lower support forging or casting in Westinghouse-designed reactors, and the lower core support beams, upper cylinder and upper core barrel flange in the core support barrel assembly in CE-designed reactors in Section 4.1.1 of this SE.

Additional expectations regarding the examination coverage and re-examination frequency are addressed in Sections 4.1.4 and 4.1.6 of this SE. **This is addressed as Topical Report Condition 1 in Section 4.1.1.**

3.2.3 Inspection of Components Subject to Irradiation-Assisted Stress Corrosion Cracking

MRP-227 grouped the following components under the "Expansion" inspection category: (1) the upper and lower core barrel girth welds and lower core barrel flange weld in Westinghouse-designed reactors; and (2) the lower cylinder girth welds in the core support barrel assembly (CSBA) in CE-designed reactors. These components were qualitatively assessed as having a "high" potential for core damage associated with their failure (i.e., they are high consequence components) and a "medium" likelihood of failure. These components were determined to be susceptible to aging effects due to SCC, IASCC and neutron embrittlement. In MRP-227, the corresponding "Primary" inspection category components were the upper core barrel flange weld in Westinghouse-designed reactors and the upper core support barrel flange weld in CE-designed reactors. These "Primary" inspection category components were judged to be most susceptible to SCC, but not susceptible to aging effects due to IASCC and neutron embrittlement.

Unlike SCC, the onset of degradation due to IASCC and neutron embrittlement depends on neutron fluence and stress levels. The incubation period for initiating cracks due to SCC is different from IASCC. Since these aging mechanisms are so different with respect to crack initiation and crack propagation, any identifiable aging effects associated with SCC in the "Primary" inspection category components may not truly represent the extent of actual aging degradation due to IASCC and neutron embrittlement in the associated "Expansion" inspection category components. Lack of any evidence of cracking due to SCC in the "Primary" inspection category components does not mean that the "Expansion" inspection category components are free of cracks due to IASCC. Therefore, the staff is concerned that the aging effects associated with IASCC and neutron embrittlement in the "Expansion" inspection category components may not be identified in a timely manner during the period of extended operation.

To ensure that the structural integrity and functionality of these high consequence of failure RVI components, which are subject to IASCC and neutron embrittlement, are maintained under all licensing basis conditions of operation during the period of extended operation, the staff has determined that the upper and lower core barrel girth welds and lower core barrel flange welds in Westinghouse-designed reactors, and the lower cylinder girth welds in the CSBA in CE-designed reactors shall be included in the "Primary" inspection category in the NRC-approved version of MRP-227. Girth welds are considered "Primary" components and axial welds are considered "Expansion" components. The examination methods shall be consistent with the MRP's recommendations addressed in MRP-227 for these components, the examination coverage for these components shall conform to the criteria described in Section 3.3.1 of this SE, and the examination frequency shall be on a 10-year interval consistent with other "Primary" inspection category components.

This is addressed as Topical Report Condition 2 in Section 4.1.2 of this SE.

3.2.4 Inspection of High Consequence Components Subject to Multiple Degradation Mechanisms

The staff evaluated the effect of multiple degradation mechanisms on the high consequence RVI components and identified that the B&W flow distributor-to-shell forging bolts and CE lower support structure core support column (casting or wrought) welds as needing to be included in the "Primary" inspection category.

B&W flow distributor-to-shell forging bolts are susceptible to SCC, fatigue, and wear. Section 3.5 of MRP-190 bases the risk band only on the single most likely aging mechanism. In Table A-1 of MRP-190 (pages A-41 and A-42), the MRP stated that SCC in the flow distributor-to-shell forging bolts is very likely to occur, whereas degradation due to fatigue and wear is less likely to occur. The safety consequence of failure of the subject component, on the other hand, is classified as "Severe" which could lead to core damage (i.e., multiple damaged fuel assemblies) with reduced margins to adequately cool the core. While SCC is regarded as the most likely degradation mechanism, the staff is concerned that the synergistic effects of SCC, fatigue and wear could potentially cause greater degradation in these bolts than just the consideration of SCC alone. Due to these synergistic effects, degradation in these bolts could then be equivalent to or greater than other components susceptible only to SCC. Therefore, the staff has concluded that B&W flow distributor-to-shell forging bolts shall be inspected as a "Primary" inspection category component.

The CE lower support structure core support column (casting or wrought) welds are susceptible to SCC, IASCC, fatigue, and irradiation embrittlement. In addition to these degradation mechanisms, this casting component is assumed to be susceptible to thermal embrittlement. These components were qualitatively assessed as having a "high" potential for core damage associated with their failure (i.e., they are high consequence components) and a "medium" likelihood of failure. MRP-232 identified IASCC and irradiation embrittlement as potential degradation mechanisms for these welds. However, the staff is concerned that the synergistic effects of SCC, fatigue, and thermal embrittlement (casting only) could potentially cause greater degradation in these welds than just the consideration of IASCC and irradiation embrittlement alone. Degradation in these welds could then be equivalent to or greater than other components susceptible only to IASCC and irradiation embrittlement due to the synergistic effects. Therefore, the staff concluded that CE lower support structure core support column (casting or wrought) welds shall be inspected as a "Primary" inspection category component.

Refer to Section 4.1.3 of this SE for administrative report change recommendations for upper core barrel (UCB) bolts, and lower grid-to-core barrel (LCB) bolts and flow distributor (FD) bolts, and their locking devices, for B&W-designed units.

The examination methods for the aforementioned components shall be consistent with the MRP's recommendations addressed in MRP-227 for these components, the examination coverage for these components shall conform to the criteria described in Sections 3.3.1 of this SE, and the examination frequency shall be on a 10-year interval consistent with other "Primary" inspection category components. **This is addressed as Topical Report Condition 3 in Section 4.1.3 in this SE.**

3.2.5 Plant-Specific Confirmation of the Applicability and Completeness of MRP-227, Revision 0

3.2.5.1 Applicability of FMECA and Functionality Analysis Assumptions

In Section 2.2 of this SE, the staff noted some of the assumptions made in the industry's FMECAs and functionality analyses. The staff questioned how it would be determined whether the operating history of a particular plant (including, for example, the effects of any plant power up-rate) was adequately represented by the assumptions made in support of the industry's FMECAs and functionality analyses. In its October 29, 2010, response to RAI 4-6 from the NRC staff's fourth set of RAIs, the MRP indicated that each applicant/licensee was responsible for assessing its plant's operating history and demonstrating that the approved version of MRP-227 is applicable to the facility. Each applicant/licensee shall refer, in particular, to the assumptions regarding plant design and operating history made in the FMECA and functionality analyses for reactors of their design (i.e., Westinghouse, CE, or B&W) which support MRP-227, and each applicant/licensee shall describe the process used for determining plant-specific differences in the design of their RVI components or plant operating conditions, which result in different component inspection categories. **This issue is Applicant/Licensee Action Item 1, and is addressed in Section 4.2.1 of this SE.**

However, the staff is also concerned that the MRP does not provide adequate guidance to allow an applicant/licensee to assess the applicability of the MRP-227 to its plant. The MRP should consider developing guidance that will allow an applicant/licensee to determine if the plant-specific differences in the design of their RVI components or plant operating conditions result in different component inspection categories. This guidance could be issued in a separate MRP report or included in a future revision of MRP-227.

3.2.5.2 PWR Vessel Internal Components Within the Scope of License Renewal

The list of RVI components for which the effects of aging will be managed by application of the AMP defined by MRP-227 is defined by Tables 4-1 and 4-2 in MRP-189, Revision 1, "Materials Reliability Program: Screening, Categorization, and Ranking of B&W-Designed PWR Internals," and Tables 4-4 and 4-5 in MRP-191.

Consistent with the requirements addressed in 10 CFR 54.4, each applicant/licensee is responsible for identifying which RVI components are within the scope of LR for its facility. Applicants/licensees shall review the information in Tables 4-1 and 4-2 in MRP-189, Revision 1, and Tables 4-4 and 4-5 in MRP-191 and identify whether these tables contain all of the RVI components that are within the scope of LR for their facilities in accordance with 10 CFR 54.4. If the tables do not identify all the RVI components that are within the scope of LR for its facility, the applicant or licensee shall identify the missing component(s) and propose any necessary modifications to the program defined in MRP-227, as modified by this SE, when submitting its plant-specific AMP such that the effects of aging on the missing component(s) will be managed for the period of extended operation. **This issue is Applicant/Licensee Action Item 2, and is addressed in Section 4.2.2 of this SE.**

3.2.5.3 Evaluation of the Adequacy of Plant-Specific Existing Programs

The MRP identified that certain CE and Westinghouse RVI components which are subject to inspection under existing programs require further plant-specific evaluation to verify the acceptability of the existing programs, or to identify changes to the existing programs which should be implemented to manage the aging of these components for the period of extended operation. If the existing programs are not acceptable, it is necessary to identify and implement changes to the programs to manage aging of applicable components over the period of extended operation. Generically, these were components for which existing plant-specific programs other than a plant's ASME Code, Section XI program were being credited for managing aging. These components were left for plant-specific evaluation because, although the MRP was able to identify that plant-specific programs already exist for the management of these components, the MRP was unable to evaluate in detail the content of each facility's plant-specific program. The CE and Westinghouse components identified for this type of plant-specific evaluation include: CE thermal shield positioning pins and CE in-core instrumentation thimble tubes (Section 4.3.2 in MRP-227), and Westinghouse guide tube support pins (split pins) (Section 4.3.3 in MRP-227). Considerations that should be included in this evaluation follow for these specific Westinghouse and CE components.

Westinghouse guide tube support pins are made from either 316 stainless steel or Alloy X750. There have been issues with cracking of the original Alloy X750 pins and many licensees have replaced them with type 316 stainless steel materials. Applicants/licensees shall evaluate the adequacy of their plant-specific existing program and ensure that the aging degradation is adequately managed during the extended period of operation for both Alloy X750 and type 316 stainless steel guide tube support pins (split pins). Therefore, it is recommended that the evaluation consider the need to replace the Alloy X750 support pins (split pins), if applicable, or inspect the replacement type 316 stainless steel support pins (split pins) to ensure that cracking has been mitigated and that aging degradation is adequately monitored during the extended period of operation.

CE fuel alignment pins are susceptible to IASCC, wear, fatigue, irradiation embrittlement, and irradiation-enhanced stress relaxation. Applicants/licensees shall evaluate the adequacy of their plant-specific existing program with respect to CE fuel alignment pins and ensure that the synergistic effects of aforementioned degradation mechanisms are adequately monitored during the extended period of operation.

Therefore, the staff determined that CE thermal shield positioning pins and in-core instrumentation thimble tubes, and Westinghouse guide tube support pins (split pins) require plant-specific evaluation to verify the acceptability of the existing programs, or to identify changes to the program that should be implemented to manage the aging of these components, for the period of extended operation. **This issue is Applicant/Licensee Action Item 3, and is addressed in Section 4.2.3 of this SE.**

3.2.5.4 B&W Core Support Structure Upper Flange Stress Relief

In its October 29, 2010, response to RAI 4-4, the MRP stated that the core support structure upper flange weld was below the screening criteria for all aging degradation mechanisms including SCC because the applied stress on this component is low and weld residual stresses have been alleviated by a stress relief heat treatment during the original fabrication. The staff accepts this technical basis, but has concluded that each applicant/licensee shall confirm the accuracy of this assumption for its facility. Therefore, B&W applicants/licensees shall confirm that the core support structure upper flange at their facilities were stress relieved during original fabrication/construction. If the upper flange weld has not been stress relieved, then this component shall be inspected as a "Primary" inspection category component consistent with the upper core support barrel weld in Westinghouse and CE units. These Westinghouse and CE components have a similar function, but have not been stress relieved.

If necessary, the examination methods and frequency for non-stress relieved B&W core support structure upper flange welds shall be consistent with the recommendations in MRP-227 for the Westinghouse and CE upper core support barrel welds. The examination coverage for this B&W flange weld shall conform to the staff's imposed criteria as described in Sections 3.3.1 and 4.3.1 of this SE.

This issue is Applicant/Licensee Action Item 4, and is addressed in Section 4.2.4 of this SE.

3.3 Evaluation of MRP-227, Revision 0 - Sections 4 and 5

The staff's review of Sections 4 and 5 of MRP-227 resulted in the staff, in principle, accepting MRP's development of I&E guidelines for the subject RVI components. The MRP considered susceptibility of RVI components to one or more degradation mechanisms and the safety consequences as a result of the failure of the RVI components in developing the I&E guidelines. However, the staff identified concerns with the MRP's proposed I&E guidelines for some components subject to MRP-227. In the following sections, the staff's evaluation of the proposed I&E guidelines for components subject to MRP-227 is provided, focusing on the staff's concerns which led to the imposition of conditions and limitations on the use of MRP-227 and plant-specific action items associated with the use of MRP-227 (as summarized in Section 4 of this SE). Additionally, in general the staff requires baseline 10-year re-examination intervals as described below.

3.3.1 General Evaluation of the MRP-227 I&E Guidelines

The staff's review of Sections 4 and 5 of MRP-227 indicated that the MRP generally provided an adequate justification regarding the examination criteria imposed for the "Primary" and "Expansion" inspection category components. "Primary" inspection category components were considered the lead components in which a degradation mechanism was expected to occur prior to the expansion components. Therefore, "Primary" inspection category components are inspected periodically. Further, the analyses indicated that "Expansion" inspection category components have a higher degree of tolerance to the aging effects to which they may be subject than their associated "Primary" inspection category components. Therefore, the initiation of inspections of "Expansion" inspection category components begins only when a particular degradation mechanism is identified in the associated "Primary" inspection category components. The staff noted that for "Primary" and "Expansion" inspection category components, the MRP generally provided examination guidelines including examination methods to be used, sampling and coverage of the examinations, expansion scope based on the extent of degradation, and evaluation of examination results for the RVI components. The staff reviewed the frequency of examinations of the RVI components addressed in tables in Section 4 of MRP-227 and concluded that, typically, the "Primary" inspection category components are to be examined during every 10-year interval.

The staff, in principle, agrees with the I&E guidelines developed for components subject to MRP-227. However in its review of the I&E guidelines, the staff identified several concerns with the MRP's proposal. Some of the staff's concerns were resolved via MRP responses to staff RAIs, and those that were not adequately resolved are reflected in plant-specific action items and/or conditions and limitations on the use of MRP-227. An example of a significant staff concern that was resolved is given in the following paragraphs, while those that were not adequately resolved are addressed in Sections 3.3.2, 3.3.3, 3.3.4, 3.3.5, 3.3.6, and 3.3.7 of this SE.

One of the staff's concerns was that, for components in the "Primary" and "Expansion" inspection categories, MRP-227 did not provide a minimum examination coverage criterion related to the total surface area/volume of the component in order to define a successful examination. The staff's concern was that, although MRP-227 states that all accessible surfaces/volumes of a component subject to inspection are to be examined, this may result in a very limited examination if plant-specific conditions limit the accessible surface area/volume.

In its October 29, 2010, response to NRC staff RAI 4-8, the MRP indicated that they will update MRP-227 to require, in addition to the requirement to examine one hundred percent of the accessible inspection area/volume for "Primary" and "Expansion" inspection category components, a minimum of 75 percent coverage of the entire examination volume (i.e., including both accessible and inaccessible regions) for all "Primary" inspection category components in order to define an inspection meeting the intent of MRP-227. For certain like-components (e.g., CE core shroud bolts) in the "Primary" inspection category, the examination "coverage" requirements are specified in terms of a minimum percentage of like components that must be inspected. In these cases, the MRP stated that the minimum sample size for inspection is 75 percent of the total population of like components. When considering the inspection of a set of like components, it is understood that essentially one hundred percent of the area/volume of each accessible like component will be examined. When accessibility is not limited, 100 percent of the specified area will be inspected.

In some cases for B&W units (table 4-1), 100 percent of the surfaces of individual welds may not be accessible. The specified visual inspection technique (VT-3) is intended to identify gross degradation in the welds and is sufficient to manage degradation of less than 100 percent accessibility of some weld surfaces. The coverage requirement for these components shall be:

- For the dowel-to-block welds: "100 percent of the accessible surfaces of 24 dowel-to-guide block welds"
- For the IMI guide tube assembly welds: "100 percent of accessible top surfaces of the 52 spider castings"

The staff has concluded that, if there are no defects discovered during the inspection, the 75 percent sample size based on inspection area/volume or total population of like components is acceptable. The staff believes that the minimum inspection area/volume or sample size is acceptable because the examined area/volume/population will provide reasonable assurance regarding the presence or absence of an active degradation mechanism in the subject component. Further, the minimum inspection area/volume is acceptable because it is assumed that the component locations that are 1) most susceptible to the degradation mechanism that is the subject of the examination and 2) most critical to component integrity will be adequately covered by the examinations as a result of the large design margins typically associated with these components. Applicants/licensees may be able to use available information to identify those specific component areas/volumes, or the subset of a group of like components, that are most likely to exhibit degradation and most important to component integrity. Using this information to prioritize the examinations will help to ensure their effectiveness.

If defects are discovered during the inspection, the licensee shall enter that information into the plant's corrective action program and evaluate whether the results of the examination ensure that the component (or set of like components) will continue to meet its intended function under all licensing basis conditions of operation until the next scheduled examination. Hence, the staff finds that the MRP has adequately addressed the staff's concern regarding a minimum examination coverage requirement for the "Primary" inspection category components.

3.3.2 Imposition of Minimum Examination Coverage Criteria for "Expansion" Inspection Category Components

In MRP-227, a requirement to examine one hundred percent of the accessible area/volume, or one hundred percent of accessible components when a population of like components (e.g., bolting) is examined, is proposed for "Expansion" inspection category components. The staff's concern is that this criterion may result in a limited examination if only a small part of a given component, or a limited number of a population of like components, is accessible for examination.

To ensure that the effects of aging are adequately monitored in the "Expansion" inspection category components, when the examination of these components is required, the staff has concluded that the minimum examination coverage requirement proposed by the MRP for

"Primary" inspection category components (discussed in Section 3.3.1 above) shall also be applied to the inspection of components in the "Expansion" inspection category. That is, a minimum of 75 percent coverage of the entire examination area or volume (i.e., including both accessible and inaccessible regions) for all "Expansion" inspection category components or a minimum sample size for inspection is 75 percent of the total population of like components will define an inspection meeting the intent of MRP-227, as approved by the NRC. For the inspection of a set of like components, it is understood that essentially 100 percent of the area/volume of each accessible like component will be examined. Application of this minimum examination coverage requirement will ensure that the inspections of "Expansion" inspection category components will be effective at identifying degradation, if present. However, applicants/licensees may also be able to use available information to identify those specific component areas/volumes, or the subset of a group of like components, that are most likely to exhibit degradation and most important to component integrity. Using this information to prioritize the examinations will help to ensure their effectiveness.

The maximum number of like components possible will be inspected (i.e., if 95 percent of the population is accessible for inspection then 95 percent must be inspected). If components have a predefined scope of inspection of less than 100 percent of accessible components, area, or length, this requirement would not apply. Examples of such exceptions include welds for which the inspection requirements call for only a certain length of weld above and below the core mid-plane to be inspected, or components where the inspection addresses a predefined sample portion of the population; for example the inspection requirements for the Westinghouse control rod guide plates (cards) which calls for a 20 percent sample inspection. Another situation where the condition would not apply is a component for which 100 percent of the population or area must be inspected and any lesser percentage of coverage would be unacceptable. This may not apply where there is a known, access limitation (generic to all plants of the NSSS type) such that the population of components or the area/volume accessible for inspection is known to be less than 75 percent of the total. However, the goal will remain that 100 percent of accessible components be inspected.

If defects are discovered during the inspection, the licensee shall enter that information into the plant's corrective action program and evaluate whether the results of the examination ensure that the component (or set of like components) will continue to meet its intended function under all licensing basis conditions of operation until the next scheduled examination.

This is addressed as Topical Report Condition 4 in Section 4.1.4 of this SE.

3.3.3 Examination Frequencies for Baffle-Former Bolts and Core Shroud Bolts

For some components, the staff was concerned over their assigned inspection frequency. For baffle-former bolts in B&W and Westinghouse-designed reactors and core shroud bolts in CE-designed reactors, the examination frequency can vary from 10 to 15 years. In Appendix B to its October 29, 2010, RAI response, the MRP indicated that the rate of radiation-induced degradation of these components may decrease in the later stage of a plant's life. The analysis that describes the reduction in the rate of degradation is described in MRP-230, "Materials Reliability Program: Functionality Analysis for Westinghouse and Combustion Engineering Representative PWR Internals." Since the rate of radiation-induced degradation may decrease in the later stage of a plant's life, the inspection interval may be able to be increased. Hence, MRP-227 provided a proposed examination frequency range of every 10 to 15 years.

Although the staff understands the general argument made in MRP-227, it has concluded that the information for the aforementioned components under the column "Examination Method/Frequency" in Tables 4-1, 4-2, and 4-3 of MRP-227 is not sufficiently prescriptive to address this issue. The entry for these components provides too much latitude with insufficient oversight of an applicant's/licensee's determination of its examination frequency. Hence, the staff determined that the NRC-approved version of MRP-227 shall specify a 10-year inspection frequency for these components following the initial or baseline inspection unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between inspections. **This is addressed as Topical Report Condition 5 in Section 4.1.5 of this SE.**

3.3.4 Periodicity of the Re-Examination of "Expansion" Inspection Category Components

The I&E guidelines for "Expansion" inspection category components are addressed in Tables 4-4, 4-5 and 4-6 in MRP-227. However, Tables 4-4, 4-5, and 4-6 in MRP-227 do not address the periodicity of subsequent re-examination for all of the "Expansion" inspection category components. For those "Expansion" inspection category components for which Tables 4-4, 4-5, and 4-6 do not specify a periodicity of subsequent re-examination, the MRP stated that the periodicity of the subsequent re-examinations depends on the results of the initial examination.

The staff has concluded that the NRC-approved version of MRP-227 shall specify a baseline periodicity of subsequent re-examination for all "Expansion" inspection category components. A baseline 10-year interval between examinations of "Expansion" inspection category components is required once degradation is identified in the associated "Primary" inspection category component. The 10-year periodicity for the "Expansion" inspection category component is applicable unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between inspections. This periodicity is consistent with ASME Code, Section XI requirements. Hence, the staff has concluded that MRP-227, Tables 4-4, 4-5, and 4-6 shall be modified to apply a baseline 10-year re-examination interval to all "Expansion" inspection category components. **This is Topical Report Condition 6, and is addressed in Section 4.1.6 of this SE.**

3.3.5 Application of Physical Measurements as Part of the I&E Guidelines for CE and Westinghouse RVI Components

The MRP proposed physical measurements as part of the I&E guidelines for some RVI components. By letter dated April 20, 2010, the MRP responded to NRC RAIs 3-11 and 3-12 and indicated that physical measurements must be utilized to monitor for loss of compressibility for Westinghouse hold down springs, and for distortion in the gap between the top and bottom core shroud segments in CE units with core barrel shrouds assembled in two vertical sections. In its response to the aforementioned RAIs, the MRP further stated that the physical measurement techniques are generally not within the scope of MRP-227, and, therefore, it did not typically provide specific acceptance criteria for these examinations.

Applicants/licensees of Westinghouse and CE plants shall identify the plant-specific acceptance criteria to be applied for their facilities when these physical examinations are made, and these acceptance criteria will be consistent with the plant's licensing basis and the need to maintain the functionality of the component being inspected under all licensing basis conditions of operation. **This is Applicant/Licensee Action Item 5, and is addressed in Section 4.2.5 of this SE.**

3.3.6 Evaluation of Inaccessible and Non-inspectable B&W Components

MRP-227 indicates that certain B&W core barrel assembly components are known to be inaccessible for inspection. They are the core barrel cylinder (including vertical and circumferential seam welds), the former plates, the external baffle-to-baffle bolts and their locking devices, and the core barrel-to-former bolts and their locking devices. The MRP also identified that B&W core barrel assembly internal baffle-to-baffle bolts are non-inspectable using currently available examination techniques. Cracking of these various components can occur due to one or more degradation mechanisms (i.e., IASCC, irradiation embrittlement, and overload). Loss of preload of the bolts can occur due to irradiation-enhanced stress relaxation and irradiation creep. Each of these components is an "Expansion" inspection category component. MRP-227 does not propose that applicants/licensees examine these inaccessible and non-inspectable components.

Applicants/licensees of B&W plants will justify the functionality of the core barrel assembly with aging degradation of these components for continued operation through the period of extended operation by performing an evaluation, or by proposing a scheduled replacement of the various components. As part of their application to implement MRP-227, applicants/licensees shall provide their justification for the continued operability of each of the inaccessible and non-inspectable components and/or provide their plan for the scheduled replacement of the components. **This is Applicant/Licensee Action Item 6, and is addressed in Section 4.2.6 of this SE.**

3.3.7 Plant-Specific Evaluation of CASS Components

The MRP identified that the following types of materials may be susceptible to reductions in their fracture toughness properties by a thermal aging embrittlement mechanism: (1) cast austenitic stainless steel (CASS) materials (the MRP cites CF8 and CF3M CASS materials); (2) martensitic stainless steel materials (the MRP cites Type 431 stainless steel); and (3) precipitation hardened stainless steel materials (the MRP cites 15-5 PH stainless steel).

In its response to RAI 4-15, dated October 29, 2010, the fourth set of RAIs, the MRP identified that some CASS RVI components require a plant-specific analysis to demonstrate that their structural integrity and functionality are maintained during the extended period of operation and that the in-core monitoring instrumentation (IMI) guide tube assembly spiders ("Primary" inspection category) and CRGT spacer castings ("Expansion" inspection category) in B&W-designed reactors, the lower support columns in CE-designed reactors ("Primary" inspection category), and lower support column bodies in Westinghouse-design reactors ("Expansion" inspection category) are examples of components that would require such a plant-specific analysis..

For B&W designs, the MRP indicated that an analysis for the B&W IMI guide tube assembly is necessary to determine the minimum number of spider arms that are needed for continued operation.

The MRP proposed to change the inspection category for B&W CRGT spacer castings from "Expansion Category" components to "Primary Category" due to the deletion of its aging management criteria for B&W vent valve disc, shaft and hinge pin components. The staff agrees with this proposed change.

With respect to the CRGT spacer castings, the response to RAI 4-15b stated, in part, that the recommended methodology of WCAP-17096 is to perform a reactivity analysis to determine the number of CRDMs that are required to shut down the reactor; and thus, no fracture mechanics evaluations are needed. However, the staff finds this approach to be unacceptable because the reactivity analysis is essentially an operability analysis of an as-found condition and does not consider the possible effect of undetected CRGT spacer cracking on the functionality of the CRDMs going forward.

Additionally, the MRP stated that a similar functionality analysis is needed for the CE lower support columns and Westinghouse lower support castings in order to demonstrate that the intended function for these components would be maintained during the extended period of operation.

The MRP's response was an attempt to address the concern of RAI 4-15 part b, which stated in part that the fracture toughness in CASS components may get so low due to thermal embrittlement and/or irradiation embrittlement and in other components due to irradiation embrittlement that preexisting fabrication or service-induced flaws that are smaller than the inspection resolution may challenge component integrity under normal loading or under design basis events. Additionally, some of these components have limited accessibility for inspection which further limits the effectiveness of inspection alone to ensure structural integrity.

To address the concerns described above, applicants/licensees shall perform a plant-specific analysis or evaluation demonstrating that the MRP-227 recommended inspections will ensure functionality of the set of components until the next scheduled inspections. Possible acceptable approaches may include, but are not limited to:

- Functionality analyses for the set of like components or assembly-level functionality analyses, or

- Component level flaw tolerance evaluation justifying that the MRP-227 recommended inspection technique(s) can detect a structurally significant flaw for the component in question, taking into account the reduction in fracture toughness due to irradiation embrittlement and thermal embrittlement; or

- For CASS, if the application of applicable screening criteria for the component's material demonstrates that the components are not susceptible to either thermal embrittlement or irradiation embrittlement, or the synergistic effects of thermal embrittlement and irradiation embrittlement combined, then no other evaluation would be necessary. For assessment of CASS materials, the licensees or applicants for LR may apply the criteria in the NRC letter of May 19, 2000, "License Renewal Issue No. 98-0030, *Thermal Aging Embrittlement of Cast Stainless Steel Components*" (NRC ADAMS Accession No. ML003717179) as the basis for determining whether the CASS materials are susceptible to the thermal aging embrittlement mechanism.

In addition, it is recommended that applicant's for LR or licensees of PWR-designed light water reactors apply these recommended actions to additional components if their IPAs confirm that the components are fabricated from those materials that the MRP has identified may be susceptible to the thermal aging embrittlement phenomenon (i.e., CASS, martensitic stainless steel or PH stainless steel materials).

The plant-specific analyses recommended in this section shall be consistent with the plant's licensing basis and should address the need to maintain the functionality of the components being evaluated under all licensing basis conditions of operation. These additional analyses should consider impacts of aging (e.g. cracking) on the intended functions of those components, or portions of components, that may not be accessible to the MRP's recommended examination technique and the possible impact that a potential loss of fracture toughness may have on the intended functions of these components as a result of both a thermal aging embrittlement mechanism and potentially a neutron irradiation embrittlement mechanism (applicable to components exposed to a high integrated neutron flux).

Therefore, applicants/licensees shall develop a plant-specific analysis to demonstrate that these components will maintain their functions during the period of extended operation. These analyses shall consider the possible loss of fracture toughness in these components due to thermal and irradiation embrittlement, and may also need to consider limitations on accessibility for inspection and the resolution/sensitivity of the inspection techniques. The plant-specific analyses shall be consistent with the plant's licensing basis and the need to maintain the functionality of the components being evaluated under all licensing basis conditions of operation. However, the requirement may not apply to components that were previously evaluated as not requiring aging management during development of MRP-227. That is, the requirement would apply to components fabricated from susceptible materials for which an individual licensee has determined aging management is required, for example during their review performed in accordance with Applicant/Licensee Action Item 2. The applicant/licensee shall include the plant-specific analysis as part of their submittal to apply the NRC-approved version of MRP-227.

This is Applicant/Licensee Action Item 7, and is addressed in Section 4.2.7 of this SE.

3.4 Evaluation of MRP-227, Revision 0 - Section 6

Section 6 of MRP-227 includes a description of the flaw evaluation methodology that is to be implemented when an examination reveals indications that do not meet acceptance criteria. Based on its review of this section, the staff concludes that this section adequately addresses, at a high level, the evaluation methodologies that could be used by the licensee or applicant for evaluating flaws detected during the examination of the RVI components. However, industry indicated in its response to RAI 4-14 that Section 6 of MRP-227 will not be used by licensees for evaluating examination results that do not meet the acceptance criteria identified in Section 5 of MRP-227. Rather, WCAP-17096-NP, Revision 2 is the document that will be used as the framework to develop those generic and plant-specific evaluations triggered by findings in the RVI examinations. The NRC staff is currently reviewing WCAP-17096-NP, Revision 2.

3.5 Evaluation of MRP-227, Revision 0 - Section 7

The staff reviewed Section 7 of MRP-227 and concludes that the implementation of MRP-227 shall comply with the implementation protocol specified in the NEI 03-08. NEI 03-08 requires that when a licensee does not implement a "Mandatory" or "Needed" element (defined in Section 2.5 of this SE) at its facility, it shall notify the NRC staff of the deviation and justification for the deviation no later than 45 days after approval by a licensee executive. Consistent with requirements addressed in Section 7.3 of MRP-227, all PWR licensees shall implement a program that is consistent with the implementation requirements addressed under the "Needed" category in NEI 03-08. Reporting of the inspection results is very essential to document the operating experience of the fleet. However, the reporting of inspection results to the industry is only addressed as a "Good Practice" element in MRP-227. Since this information will be used to update the I&E guidelines and to inform subsequent examinations at nuclear power plants, the staff recommends that reporting of inspection results be classified under the "Needed" category.

3.5.1 Submittal of Information for Staff Review and Approval

In addition to the implementation of MRP-227 in accordance with NEI 03-08, applicants/licensees whose licensing basis contains a commitment to submit a PWR RVI AMP and/or inspection program shall also make a submittal for NRC review and approval to credit their implementation of MRP-227, as amended by this SE. An applicant's/licensee's application to implement MRP-227, as amended by this SE shall include the following items (1) and (2). Applicants who submit applications for LR after the issuance of this SE shall, in accordance with the NUREG-1801, Revision 2, submit the information provided in the following items (1) through (5) for staff review and approval.

1. An AMP for the facility that addresses the 10 program elements as defined in NUREG-1801, Revision 2, AMP XI.M16A.

2. To ensure the MRP-227 program and the plant-specific action items will be carried out by applicants/licensees, applicants/licensees are to submit an inspection plan which addresses the identified plant-specific action items for staff review and approval consistent with the licensing basis for the plant. If an applicant/licensee plans to implement an AMP which deviates from the guidance provided in MRP-227, as approved by the NRC, the applicant/licensee shall identify where their program deviates from the recommendations of MRP-227, as approved by the NRC, and shall provide a justification for any deviation which includes a consideration of how the deviation affects both "Primary" and "Expansion" inspection category components.

3. The regulation at 10 CFR 54.21(d) requires that an FSAR supplement for the facility contain a summary description of the programs and activities for managing the effects of aging and the evaluation of TLAAs for the period of extended operation. Those applicants for LR referencing MRP-227, as approved by the NRC, for their RVI component AMP shall ensure that the programs and activities specified as necessary in MRP-227, as approved by the NRC, are summarily described in the FSAR supplement.

4. The regulation at 10 CFR 54.22 requires each applicant for LR to submit any TS changes (and the justification for the changes) that are necessary to manage the effects

of aging during the period of extended operation as part of its LR application (LRA). For the plant CLBs that include mandated inspection or analysis requirements for RVI either in the operating license for the facility or in the facility TS, the applicant/licensee shall compare the mandated requirements with the recommendations in the NRC-approved version of MRP-227. If the mandated requirements differ from the recommended criteria in MRP-227, as approved by the NRC, the conditions in the applicable license conditions or TS requirements take precedence over the MRP recommendations and shall be complied with.

5. Pursuant to 10 CFR 54.21(c)(1), the applicant is required to identify all analyses in the CLB for their RVI components that conform to the definition of a TLAA in 10 CFR 54.3 and shall identify these analyses as TLAAs for the application in accordance with the TLAA identification requirement in 10 CFR 54.21(c)(1). MRP-227 does not specifically address the resolution of TLAAs that may apply to applicant/licensee RVI components. Hence, applicants/licensees who implement MRP-227, as approved by the NRC, shall still evaluate the CLB for their facilities to determine if they have plant-specific TLAAs that shall be addressed. If so, the applicant's/licensee's TLAA shall be submitted for NRC review along with the applicant's/licensee's application to implement the NRC-approved version of MRP-227.

   For those cumulative usage factor (CUF) analyses that are TLAAs, the applicant may use the PWR Vessel Internals Program as the basis for accepting these CUF analyses in accordance with 10 CFR 54.21(c)(1)(iii) only if the RVI components in the CUF analyses are periodically inspected for fatigue-induced cracking in the components during the period of extended operation. The periodicity of the inspections of these components shall be justified to be adequate to resolve the TLAA. Otherwise, acceptance of these TLAAs shall be done in accordance with either 10 CFR 54.21(c)(1)(i) or (ii), or in accordance with 10 CFR 54.21(c)(1)(iii) using the applicant's program that corresponds to NUREG-1801, Revision 2, AMP X.M1, "Metal Fatigue of Reactor Coolant Pressure Boundary Program". To satisfy the evaluation requirements of ASME Code, Section III, Subsection NG-2160 and NG-3121, the existing fatigue CUF analyses shall include the effects of the reactor coolant system water environment.

This is Applicant/Licensee Action Item 8, and is addressed in Section 4.2.8 of this SE.

3.6 Evaluation of MRP-227 - Appendix A

The staff reviewed Appendix A of MRP-227 which originally addressed 3 of the 10 program attributes of an AMP. The staff noted that discussion of the three AMP attributes in MRP-227, Appendix A did not entirely conform to the NRC's recommended program element criteria for AMPs that are given in Section A.1.2.3 of NRC Branch Technical Position RLSB-1. In the MRP response to RAI Set 4, the MRP stated that Appendix A in MRP-227 would be deleted entirely from the scope of MRP-227 and replaced with a new Appendix A entitled Operating Experience Summary.

It was the staff's intent to use the information provided in MRP-227, Appendix A to develop Revision 2 of NUREG-1801, AMP XI.M16A, "PWR Vessel Internals Program." By letter dated November 12, 2009, the staff requested that the MRP provide additional information in a format that conforms to the recommended program element criteria in Section A.1.2.3 of NRC Branch Technical Position RLSB-1 that could be used to develop NUREG-1801, Revision 2, AMP XI.M16A and that could be adopted for the contents of an applicant's PWR RVI AMP. By letter dated December 2, 2009, the MRP provided a revised AMP that the MRP recommended for the development of the NUREG-1801, Revision 2. AMP XI.M16A in NUREG-1801, Revision 2 (or in subsequent revisions of NUREG-1801 that follow) is the staff's recommended AMP for PWR RVI components.

When the approved version of MRP-227 is published, MRP-227, Appendix A shall be updated to include a reference to AMP XI.M16A, in NUREG-1801, Revision 2 (or in subsequent revisions of the GALL report that follow) and the Operating Experience Summary that is mentioned in the MRP's response to RAI Set 4. **This is addressed as Topical Report Condition 7 in Section 4.1.7 of this SE.**

3.7 Changes to MRP Recommended "Primary Category" Inspection Activities for B&W-Design Vent Valve Components and Their "Expansion Category" Component Links

During a conference call held with the MRP on October 27, 2011, a B&W participant in the MRP process commented that the discs, shafts, and hinge pins in B&W-design vent valve designs should not be subject to aging management under the requirements of 10 CFR Part 54. The staff confirmed to the MRP that the requirements in 10 CFR 54.21(a)(1) do not require a component at a facility to be subject to an AMR if either the component involves a movable part or a change in configuration (i.e., active RVI components) or if the component would be subject to replacement activities based on a qualified life or specified time period (i.e., consumable items). Therefore, the staff supports the rationale for removing the MRP's "Primary Category" aging management recommendations for the B&W vent valve discs, shafts, and hinge pins from the scope of MRP-227, and that the MRP could remove the aging management recommendations for the vent valve discs, shafts, and hinge pins from the scope of MRP-227.

However, the staff also informed the MRP that the CRGT spacer castings would then need to be identified as B&W-design "Primary Category" components because the spacer castings were categorized previously as "Expansion" category components, the associated "Primary" category component for which was the vent valve discs. Since the vent valve discs would no longer be cross referencing to the spacer castings as "Expansion Category" component links, the MRP proposed a change to the inspection category criterion for the CRGT spacer castings from "Expansion Category" components to "Primary Category" components for the MRP's I&E methodology and indicated that the components would be inspected on a 10-year frequency using a VT-3 visual examination technique. The staff finds the change acceptable since it will ensure that the CASS components most susceptible to TE will be inspected. Consistent with SE Sections 3.3.7 and 4.2.7, the CRGT spacer casting are subject to the Applicant/Licensee Action Item No. 7 on the MRP-227 methodology.

In Table 4-1 of MRP-227, the MRP identified that the pads, pad-to-rib section welds, and Alloy X-750 dowel cap screws and their locking devices in the lower grid fuel assembly were applicable "Expansion" category components for the Primary inspections that would be performed on the CRGT spacer castings. The relevant mechanism for the spacer castings is thermal embrittlement, while the relevant mechanism for the pads, pad-to-rib section welds, and

Alloy X-750 dowel cap screws and their locking devices in the lower grid fuel assembly is irradiation embrittlement.

These changes are part of **Topical Report Condition Item 3** (Refer to SE Section 4.1.3).

4.0 CONDITIONS AND LIMITATIONS AND APPLICANT/LICENSEE PLANT-SPECIFIC ACTION ITEMS

Based on its review, the NRC staff identified some issues and concerns in Section 3.0 of this SE that were not adequately resolved regarding the implementation of MRP-227. Some of the staff's issues that are not adequately resolved and remaining concerns are related to conditions and limitations on the use of MRP-227. These conditions and limitations address deficiencies in the AMP defined by MRP-227 and are identified in Section 4.1 of this SE. In addition, some of the staff's issues and concerns that were not adequately resolved are related to applicant/licensee action items related to the use of MRP-227. These plant-specific action items address topics related to the implementation of MRP-227 that could not be effectively addressed on a generic basis in MRP-227 and are identified in Section 4.2 of this SE. Although Section 4.1 and 4.2 describe the conditions and limitations and the plant-specific action items identified by the NRC staff, Section 3 more fully describes all concerns and shall be considered during any update to MRP-227 to comport with this SE. In addition, the re-examination frequency for "Primary" inspection category components shall be on a maximum 10-year interval, unless a plant-specific analysis providing justification for an extended examination frequency is submitted to and approved by the NRC.

4.1 Limitations and Conditions on the Use of MRP-227

4.1.1 High Consequence Components in the "No Additional Measures" Inspection Category

As discussed in Section 3.2.2 of this SE, the staff determined that certain high consequence of failure components were binned in the MRP-227 "No Additional Measures" inspection category. To ensure that the structural integrity and functionality of these RVI components are maintained under all licensing basis conditions during the period of extended operation, the staff has determined that each of these components shall be included in the "Expansion" inspection category in the NRC-approved version of MRP-227. The examination method to be used for these additional "Expansion" inspection category components shall be consistent with the examination method for the "Primary" inspection category component to which they are linked. The "Primary" inspection category components to which these additional "Expansion" inspection category components shall be linked is shown below.

Component	Link to "Primary" Inspection Category Components
Upper core plate in Westinghouse-designed reactors	CRGT lower flange weld
Lower support forging or casting in Westinghouse-designed reactors	CRGT lower flange weld

Lower core support beams in CE-designed reactors	Upper core support barrel flange weld
Core support barrel assembly upper cylinder and upper core barrel flange in CE-designed reactors	Upper core support barrel flange weld

The examination coverage and re-examination frequency requirements for these "Expansion" inspection category components shall be as addressed in Sections 4.1.4 and 4.1.6 of this SE.

When publishing the approved version of MRP-227, Tables 4-5, and 4-6 shall be revised accordingly. **This is Topical Report Condition 1.**

4.1.2 Inspection of Components Subject to Irradiation-Assisted Stress Corrosion Cracking

As discussed in Section 3.2.3 of this SE, the staff noted that there are inconsistencies between the degradation mechanisms between some of the "Primary" and associated "Expansion" inspection category components in Westinghouse and CE-designed reactors. The MRP identified IASCC and neutron embrittlement as the degradation mechanisms for the following "Expansion" inspection category components, whereas SCC was identified as the degradation mechanism for the corresponding "Primary" inspection category components. The following table identifies the subject "Expansion" inspection category components and their corresponding tables from MRP-227.

"Expansion" Inspection Category Components Subject to IASCC	**Tables in MRP-227, Revision 0**
Upper and lower core barrel cylinder girth welds in Westinghouse-designed reactors	Table 4-6
Lower core barrel flange weld in Westinghouse-designed reactors	Table 4-6
Core support barrel assembly lower cylinder girth welds in CE-designed reactors	Table 4-5

To ensure that the structural integrity and functionality of these RVI components are maintained under all licensing basis conditions during the period of extended operation, the staff has determined that each of these components shall be included in the "Primary" inspection category in the NRC-approved version of MRP-227. The examination methods shall be consistent with the MRP's recommendations for these components, the examination coverage for these components shall conform to the criteria described in Section 3.3.1 of this SE, and the re-examination frequency shall be on a 10-year interval consistent with other "Primary" inspection category components. For both Westinghouse and CE designed reactors, girth welds are normally considered Primary components and the axial welds are normally considered Expansion components. For Westinghouse and CE designed reactors, the inspection shall be expanded to axial welds (expansion component) in the event that degradation is observed in the girth welds.

When publishing the approved version of MRP-227, Revision 0 Tables 4-2 and 4-3 shall be revised accordingly. **This is Topical Report Condition 2.**

4.1.3 Inspection of High Consequence Components Subject to Multiple Degradation Mechanisms

As discussed in Section 3.2.4 of this SE, the NRC staff determined that two high-consequence of failure components subject to important combinations of multiple degradation mechanisms were binned in the MRP-227 "Expansion" inspection category. The following table includes the identification of these components and their corresponding tables from MRP-227.

Component	**Relevant Table**
Flow distributor-to-shell forging bolts in B&W-designed reactors	Table 4-4
Core support column (casting or wrought) welds in lower support structure in CE-designed reactors	Table 4-5

To ensure that the structural integrity and functionality of these RVI components are maintained under transient loading conditions during the period of extended operation, the staff has determined that the subject components shall be included in the "Primary" inspection category in the NRC-approved version of MRP-227. The examination methods shall be consistent with the MRP's recommendations for these components, the examination coverage for the aforementioned components shall conform to the criteria as described in Section 3.3.1 of this SE, and the re-examination frequency shall be on a 10-year interval similar to other "Primary" inspection category components.

In addition, in MRP-227 Table 4-1, the MRP included the B&W LCB bolts, flow distributor (FD) bolts, and their locking devices as applicable "Expansion Category" components for B&W upper core barrel (UCB) bolts and their locking devices. Note 3 indicated this expansion was only applicable if the primary inspections of the LCB bolts or FD bolts had not yet been conducted. However, since the LCB and FD bolts are already included as "Primary" inspection category components in MRP-227 Table 4-1, it is inappropriate to include the LCB and FD bolts in the "Expansion Link" column references. Specifically, the staff determined that this note could be interpreted to mean that B&W licensees would not need to perform the primary inspections of the LCB or FD bolts at their next scheduled opportunity because, as "Expansion" components, the inspections of LCB or FD bolts would only be performed if aging was detected from the "Primary" inspections of the UCB bolts. Therefore, the staff requires the MRP to move the reference for Note 3 to the "Examination Method/Frequency" column entry for the LCB bolts and FD bolts in MRP-227 Table 4-1 and should delete the "Expansion Category" reference for the LCB bolts and FD bolts from the "Expansion Link" column of the UCB bolt line item in that table.

In Table 4-1 of MRP-227, the MRP identified that the pads, pad-to-rib section welds, and Alloy X-750 dowel cap screws and their locking devices in the lower grid fuel assembly were applicable "Expansion" category components for the Primary inspections that would be performed on the CRGT spacer castings. The relevant mechanism for the spacer castings is thermal embrittlement, while the relevant mechanism for the pads, pad-to-rib section welds, and Alloy X-750 dowel cap screws and their locking devices in the lower grid fuel assembly is irradiation embrittlement. Consistent with SE Sections 3.3.7 and 4.2.7, the CRGT spacer castings are subject to the Applicant/Licensee Action Item No. 7 on the MRP-227 methodology.

When publishing the approved version of MRP-227, Tables 3-1, 4-1, 4-2, 4-4, and 4-5 shall be revised accordingly. **This is Topical Report Condition 3.**

4.1.4 Imposition of Minimum Examination Coverage Criteria for "Expansion" Inspection Category Components

As discussed in Section 3.3.1 and Section 3.3.2 of this SE, for "Primary" inspection category components, MRP-227 will require that 100 percent of a "Primary" inspection category component's accessible inspection area or volume be examined and 75 percent of a "Primary" inspection category component's total (accessible + inaccessible) inspection area or volume be examined or, when addressing a set of like components (e.g., bolting), that the inspection examine a minimum sample size of 75 percent of the total population of like components. For the inspection of a set of like components, 100 percent of the accessible volume/area of each accessible like component will be examined. This defines the minimum inspection required to meet the intent of MRP-227 provided that no defects are discovered during the inspection. If defects are discovered during the inspection, the licensee shall enter that information into the plant's corrective action program and evaluate whether the results of the examination ensure that the component (or set of like components) will continue to meet its intended function under all licensing basis conditions of operation until the next scheduled examination.

The maximum number of like components possible will be inspected (i.e., if 95 percent of the population is accessible for inspection then 95 percent must be inspected). This condition does not apply to components having a predefined scope of inspection less than 100 percent of accessible components, area, or length. Examples of such exceptions include welds for which the inspection requirements call for only a certain length of weld above and below the core mid-plane to be inspected, or components where the inspection addresses a predefined sample portion of the population; for example the inspection requirements for the Westinghouse control rod guide plates (cards) which calls for a 20 percent sample inspection. Another situation where the condition would not apply is a component for which 100 percent of the population or area must be inspected and any lesser percentage of coverage would be unacceptable. This condition also is understood not to apply where there is a known access limitation (generic to all plants of the NSSS type) such that the population of components or the area/volume accessible for inspection is known to be less than 75 percent of the total.

As discussed in Section 3.3.2 of this SE, an equivalent requirement shall be imposed for the inspection of components in the MRP-227 "Expansion" inspection category.

When the approved version of MRP-227 is published, Tables 4-4, 4-5, and 4-6 shall be updated to include this requirement. **This is Topical Report Condition 4.**

4.1.5 Examination Frequencies for Baffle-Former Bolts and Core Shroud Bolts

As discussed in Section 3.3.3 of this SE, Tables 4-1, 4-2, and 4-3 of MRP-227 indicate that the frequency of examinations for the baffle-former bolts of B&W and Westinghouse-designed reactors and core shroud bolts in CE-designed reactors can vary from 10 to 15 years. However, the staff notes that MRP-227 provides too much latitude with insufficient oversight of an applicant's/licensee's determination of its examination frequency. Hence, the staff has determined that the NRC-approved version of MRP-227 shall specify a 10-year inspection frequency for these components following the initial or baseline inspection unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between inspections. MRP-227 Tables 4-1, 4-2, and 4-3 shall be modified when the approved version of MRP-227 is published to reflect this change. **This is Topical Report Condition 5.**

4.1.6 Periodicity of the Re-examination of "Expansion" Inspection Category Components

As discussed in Section 3.3.4 of this SE, MRP-227 Tables 4-4, 4-5, and 4-6 shall be modified when the approved version of MRP-227 is published to apply a baseline 10-year re-examination interval to all "Expansion" inspection category components (once degradation is identified in the associated "Primary" inspection category component and examination of the "Expansion" category component commences) unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between inspections. **This is Topical Report Condition 6.**

4.1.7 Updating of MRP-227 Appendix A

As discussed in Section 3.6 of this SE, when the approved version of MRP-227 is published, MRP-227, Appendix A shall be updated to include a reference to AMP XI.M16A in NUREG-1801, Revision 2 (or in subsequent revisions of the GALL report that follow) and the Operating Experience Summary. **This is Topical Report Condition 7.**

4.2 Plant-Specific Action Items

4.2.1 Applicability of FMECA and Functionality Analysis Assumptions

As addressed in Section 3.2.5.1 of this SE, each applicant/licensee is responsible for assessing its plant's design and operating history and demonstrating that the approved version of MRP-227 is applicable to the facility. Each applicant/licensee shall refer, in particular, to the assumptions regarding plant design and operating history made in the FMECA and functionality analyses for reactors of their design (i.e., Westinghouse, CE, or B&W) which support MRP-227 and describe the process used for determining plant-specific differences in the design of their RVI components or plant operating conditions, which result in different component inspection categories. The applicant/licensee shall submit this evaluation for NRC review and approval as part of its application to implement the approved version of MRP-227. **This is Applicant/Licensee Action Item 1.**

4.2.2 PWR Vessel Internal Components Within the Scope of License Renewal

As discussed in Section 3.2.5.2 of this SE, consistent with the requirements addressed in 10 CFR 54.4, each applicant/licensee is responsible for identifying which RVI components are within the scope of LR for its facility. Applicants/licensees shall review the information in Tables 4-1 and 4-2 in MRP-189, Revision 1, and Tables 4-4 and 4-5 in MRP-191 and identify whether these tables contain all of the RVI components that are within the scope of LR for their facilities in accordance with 10 CFR 54.4. If the tables do not identify all the RVI components that are within the scope of LR for its facility, the applicant or licensee shall identify the missing component(s) and propose any necessary modifications to the program defined in MRP-227, as modified by this SE, when submitting its plant-specific AMP. The AMP shall provide assurance that the effects of aging on the missing component(s) will be managed for the period of extended operation. **This issue is Applicant/Licensee Action Item 2.**

4.2.3 Evaluation of the Adequacy of Plant-Specific Existing Programs

As addressed in Section 3.2.5.3 in this SE, applicants/licensees of CE and Westinghouse are required to perform plant-specific analysis either to justify the acceptability of an applicant's/licensee's existing programs, or to identify changes to the programs that should be implemented to manage the aging of these components for the period of extended operation. The results of this plant-specific analyses and a description of the plant-specific programs being relied on to manage aging of these components shall be submitted as part of the applicant's/licensee's AMP application. The CE and Westinghouse components identified for this type of plant-specific evaluation include: CE thermal shield positioning pins and CE in-core instrumentation thimble tubes (Section 4.3.2 in MRP-227), and Westinghouse guide tube support pins (split pins) (Section 4.3.3 in MRP-227). **This is Applicant/Licensee Action Item 3.**

4.2.4 B&W Core Support Structure Upper Flange Stress Relief

As discussed in Section 3.2.5.4 of this SE, the B&W applicants/licensees shall confirm that the core support structure upper flange weld was stress relieved during the original fabrication of the Reactor Pressure Vessel in order to confirm the applicability of MRP-227, as approved by the NRC, to their facility. If the upper flange weld has not been stress relieved, then this component shall be inspected as a "Primary" inspection category component. If necessary, the examination methods and frequency for non-stress relieved B&W core support structure upper flange welds shall be consistent with the recommendations in MRP-227, as approved by the NRC, for the Westinghouse and CE upper core support barrel welds. The examination coverage for this B&W flange weld shall conform to the staff's imposed criteria as described in Sections 3.3.1 and 4.3.1 of this SE. The applicant's/licensee's resolution of this plant-specific action item shall be submitted to the NRC for review and approval. **This is Applicant/Licensee Action Item 4.**

4.2.5 Application of Physical Measurements as part of I&E Guidelines for B&W, CE, and Westinghouse RVI Components

As addressed in Section 3.3.5 in this SE, applicants/licensees shall identify plant-specific acceptance criteria to be applied when performing the physical measurements required by the NRC-approved version of MRP-227 for loss of compressibility for Westinghouse hold down springs, and for distortion in the gap between the top and bottom core shroud segments in CE units with core barrel shrouds assembled in two vertical sections. The applicant/licensee shall include its proposed acceptance criteria and an explanation of how the proposed acceptance criteria are consistent with the plants' licensing basis and the need to maintain the functionality of the component being inspected under all licensing basis conditions of operation during the period of extended operation as part of their submittal to apply the approved version of MRP-227. **This is Applicant/Licensee Action Item 5.**

4.2.6 Evaluation of Inaccessible B&W Components

As addressed in Section 3.3.6 in this SE, MRP-227 does not propose to inspect the following inaccessible components: the B&W core barrel cylinders (including vertical and circumferential seam welds), B&W former plates, B&W external baffle-to-baffle bolts and their locking devices, B&W core barrel-to-former bolts and their locking devices, and B&W core barrel assembly internal baffle-to-baffle bolts. The MRP also identified that although the B&W core barrel assembly internal baffle-to-baffle bolts are accessible, the bolts are non-inspectable using currently available examination techniques.

Applicants/licensees shall justify the acceptability of these components for continued operation through the period of extended operation by performing an evaluation, or by proposing a scheduled replacement of the components. As part of their application to implement the approved version of MRP-227, applicants/licensees shall provide their justification for the continued operability of each of the inaccessible components and, if necessary, provide their plan for the replacement of the components for NRC review and approval. **This is Applicant/Licensee Action Item 6.**

4.2.7 Plant-Specific Evaluation of CASS Materials

As discussed in Section 3.3.7 of this SE, the applicants/licensees of B&W, CE, and Westinghouse reactors are required to develop plant-specific analyses to be applied for their facilities to demonstrate that B&W IMI guide tube assembly spiders and CRGT spacer castings, CE lower support columns, and Westinghouse lower support column bodies will maintain their functionality during the period of extended operation or for additional RVI components that may be fabricated from CASS, martensitic stainless steel or precipitation hardened stainless steel materials. These analyses shall also consider the possible loss of fracture toughness in these components due to thermal and irradiation embrittlement, and may also need to consider limitations on accessibility for inspection and the resolution/sensitivity of the inspection techniques. The requirement may not apply to components that were previously evaluated as not requiring aging management during development of MRP-227. That is, the requirement would apply to components fabricated from susceptible materials for which an individual licensee has determined aging management is required, for example during their review performed in accordance with Applicant/Licensee Action Item 2. The plant-specific analysis shall be consistent with the plant's licensing basis and the need to maintain the functionality of the components being evaluated under all licensing basis conditions of operation. The applicant/licensee shall include the plant-specific analysis as part of their submittal to apply the approved version of MRP-227. **This is Applicant/Licensee Action Item 7.**

4.2.8 Submittal of Information for Staff Review and Approval

As addressed in Section 3.5.1 in this SE, applicants/licensees shall make a submittal for NRC review and approval to credit their implementation of MRP-227, as amended by this SE, as an AMP for the RVI components at their facility. This submittal shall include the information identified in Section 3.5.1 of this SE. **This is Applicant/Licensee Action Item 8.**

5.0 CONCLUSIONS

The staff has reviewed MRP-227, Revision 0 and concludes that MRP-227, as modified by the conditions and limitations and applicant/licensee action items summarized in Section 4.0 of this SE, provides for the development of an AMP for PWR RVI components within the scope of MRP-227 which will adequately manage their aging effects such that there is reasonable assurance that they will perform their intended functions in accordance with the CLB during the extended period of operation.

Any applicant may reference MRP-227 as modified by this SE and approved by the NRC, in a LRA or other licensing action to satisfy the requirements of 10 CFR 54.21(a)(3) for demonstrating that the effects of aging on the RVI components, within the scope of MRP-227, will be adequately managed. The staff also concludes that, upon completion of plant-specific action items set forth in Section 4.0, referencing the NRC-approved version of MRP-227 in a LRA and summarizing the AMP contained in MRP-227 in a FSAR supplement will provide the staff with sufficient information to make necessary findings required by 10 CFR 54.29(a)(1) for RVI components within the scope of MRP-227, as approved by the NRC.

6.0 REFERENCES

The following MRP reports and supporting information were used by the staff as part of its review of the MRP-227.

1. NUREG-1801 Revision 2, "Generic Aging Lessons Learned (GALL)."

2. MRP-175 Revision 0, "Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values," ADAMS Accession Number ML063470637.

3. MRP-189 Revision 1, "Materials Reliability Program: Screening, Categorization, and Ranking of B& W-Designed PWR Internals," ADAMS Accession Number ML092250189.

4. MRP-190 Revision 0, "Materials Reliability Program: Failure Modes, Effects, and Criticality Analysis of B&W-Designed PWR Internals," ADAMS Accession Number ML091910128.

5. MRP-191 Revision 0, "Materials Reliability Program: Screening, Categorization and Ranking of Reactor Internals of Westinghouse and Combustion Engineering PWR Designs," ADAMS Accession Number ML091910130.

6. MRP-210 Revision 0, "Materials Reliability Program: Fracture Toughness Evaluation of Highly Irradiated PWR Stainless Steel Internal Components," ADAMS Accession Number ML092230736.

7. MRP-211 Revision 0, "Materials Reliability Program: PWR Internals: Age Related Material Properties Degradation Mechanisms, Models and Basis Data," ADAMS Accession Number ML093020614.

8. MRP-228 Revision 0, "Materials Reliability Program: Inspection Standard for PWR Internals," ADAMS Accession Number ML092120574.

9. MRP-229 Revision 3, "Materials Reliability Program: Functionality Analysis for B& W Representative PWR Internals," ADAMS Accession Numbers ML110280110, ML110280111, and ML110280112.

10. MRP-230 Revision 1," Materials Reliability Program: Functionality Analysis for Westinghouse and Combustion Engineering Representative PWR Internals," ADAMS Accession Numbers ML093210269, ML093210270, and ML093210271.

11. MRP-231 Revision 2, "Materials Reliability Program: Aging Management Strategies for B& WPWR Internals," ADAMS Accession Number ML110280113.

12. MRP-232 Revision 0, "Materials Reliability Program: Aging Management Strategies for Westinghouse and Combustion Engineering PWR Internals," ADAMS Accession Numbers ML091671780, ML092250192, and ML092230745.

13. MRP-276 Revision 0, "Materials Reliability Program: Thermal Aging and Neutron Embrittlement Assessment of Cast Austenitic Stainless Steel Welds in PWR Internals" ADAMS Accession Number ML102950165.

14. WCAP-17096-NP, Revision 2, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," dated December 2009, ADAMS Accession Number ML101460157.

15. Materials Reliability Program (MRP) Report 1016596 (MRP-227), Revision 0, "Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines, dated January 12, 2009, ADAMS Accession No. ML090160204.

16. Response to the staff RAIs dated August 24, 2009, ADAMS Accession Number ML092870179.

17. Response to the staff RAIs dated November 12, 2009 ADAMS Accession Number ML101120660.

18. Response to the staff RAIs dated September 30, 2010, ADAMS Accession Number ML103160381.

19. Safety Evaluation, Revision 0 to TR MRP-227, dated June 22, 2011, ADAMS Accession No. ML111600498.

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CITATIONS

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REPORT SUMMARY

The Materials Reliability Program (MRP) developed inspection and evaluation (I&E) guidelines for managing long-term aging reactor vessel internal components of pressurized water reactors (PWRs) reactor internals. Specifically, the guidelines are applicable to reactor vessel internal structural components; they do not address fuel assemblies, reactivity control assemblies, or welded attachments to the reactor vessel.

Background
Demonstrating that effects of aging degradation in PWR internals are adequately managed is essential for maintaining a healthy fleet and assuring continued functionality of reactor internals. As a work product of the MRP, these I&E guidelines are intended to support that demonstration, with requirements for inspections to detect effects of aging degradation. The program to develop these guidelines has been underway for almost a decade, organized around a framework and strategy for managing effects of aging in PWR internals, dependent on a substantial database of material data and supporting evaluation results. The goal of this development was primarily to support license renewal, but the guidelines are intended to apply to the current license period as well.

Objectives
To provide generic I&E guidelines for each PWR design for use by individual plant owners to develop engineering programs to manage aging in Pressurized Water Reactor (PWR) internals. It is also intended to support the industry in preparing and executing their PWR internals aging management programs (AMPs) needed to satisfy license renewal commitments.

Approach
An experienced team consisting of utility and nuclear steam supply system (NSSS) vendors and EPRI experts, representing a broad spectrum of reactor design, operations, and materials expertise, worked on the project. The team reviewed available data and industry experience on materials aging to develop a systematic approach for identifying and prioritizing inspection requirements for internals. The key sequential steps in the process included the following:

1. development of screening criteria, with susceptibility levels for the eight postulated aging mechanisms relevant to reactor internals and their effects;
2. initial component screening and categorization, using susceptibility levels and FMECA (failure modes, effects, and criticality analysis) to identify the relative ranking of components;
3. functionality assessment of degradation for components and assemblies of components; and
4. aging management strategy development combining results of the functionality assessment with component accessibility, operating experience, existing

evaluations, and prior examination results to determine the appropriate aging management methodology, baseline examination timing, and the need for and the timing of subsequent inspections.

Through this process, reactor internals for all three PWR designs were evaluated, and appropriate recommendations for aging management actions specific to each component were provided.

Results

One "mandatory" and five "needed" implementation requirements have been developed. These requirements provide the framework and details for individual utility engineering programs for managing aging in reactor internal components and the development of AMPs to support license renewal.

EPRI Perspective

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. fleet of PWRs. The aging management strategies reports (MRP-231 and MRP-232) provide the basis for these guidelines. The functional evaluations that support the guidelines were based on representative configurations and operational histories, which were generally conservative, but not necessarily bounding in every parameter. These assumptions are a conservative representation of U.S. PWR operating plants, all of which implemented low-leakage core-loading patterns early in their 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 also are considered applicable to plants that have replaced components or component assemblies; however, alternatives can be technically justified.

The Inspection Standard for PWR internals (MRP-228) is the companion document to these I&E guidelines and provides examination requirement standards for components listed in the guidelines.

Keywords

Pressurized water reactor
Reactor internals
Inspection guidelines
Aging management
License renewal
Material reliability program

LIST OF ACRONYMS

AMP	Aging Management Program
ASME	American Society of Mechanical Engineers
B&PV	Boiler & Pressure Vessel
B&W	Babcock & Wilcox
BB	Baffle-to-Baffle
BMI	Bottom Mounted Instrumentation
BWR	Boiling Water Reactor
BWRVIP	Boiling Water Reactor Vessel & Internals Project
CAP	Corrective Action Program
CASS	Cast Austenitic Stainless Steel
CB	Core Barrel
CBF	Core Barrel-to-Former
CE	Combustion Engineering
CEA	Control Element Assembly
CFR	Code of Federal Regulations
CR-3	Crystal River Unit 3
CRGT	Control Rod Guide Tube
CSA	Core Support Assembly
CSS	Core Support Shield
DB	Davis-Besse
E	Expansion, I&E Guidelines Component Group
ECP	Electro-Chemical Potential
EFPY	Effective Full Power Years
EPFM	Elastic-Plastic Fracture Mechanics
EPRI	Electric Power Research Institute
ET	Electromagnetic Testing (Eddy Current)
EVT	Enhanced Visual Testing (a Visual NDE Method that includes EVT-1)
FB	Baffle-to-Former
FD	Flow Distributor
FMECA	Failure Mode, Effects, and Criticality Analysis
GALL	Generic Aging Lessons Learned
HWC	Hydrogen Water Chemistry

I&E	Inspection and Evaluation
IASCC	Irradiation-Assisted Stress Corrosion Cracking
ICI	In-Core Instrumentation
IGSCC	Intergranular SCC
IMI	Incore Monitoring Instrumentation
IP	Issue Program
ISI	Inservice Inspection
ISR	Irradiation-Enhanced Stress Relaxation
ITG	Issue Task Group
JOBB	Joint Owners Baffle Bolt
LCB	Lower Core Barrel
LCP	Lower Core Plate
LEFM	Linear Elastic Fracture Mechanics
LTS	Lower Thermal Shield
LOCA	Loss-of-Coolant-Accident
MRP	Materials Reliability Program
N	No Additional Measures, I&E Guidelines Component Group
NDE	Non-Destructive Examination
NEI	Nuclear Energy Institute
NRC	U. S. Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OBE	Operating Basis Earthquake
ONS	Oconee Nuclear Station (ONS-1, ONS-2, and ONS-3)
P	Primary, I&E Guidelines Component Group
PH	Precipitation-Hardenable (Heat Treatment)
PMMP	Preventive Maintenance Management Program
PWR	Pressurized Water Reactor
PWROG	Pressurized Water Reactor Owners Group
PWSCC	Primary Water SCC
QA	Quality Assurance
RCS	Reactor Coolant System
RI-FG	Reactor Internals Focus Group
RI-ITG	Reactor Internals Issue Task Group
SCC	Stress Corrosion Cracking
SS	Stainless Steel
SSE	Safe Shutdown Earthquake
SSHT	Surveillance Specimen Holder Tube
TLAA	Time-Limited Aging Analysis
TMI-1	Three Mile Island Unit 1

UCB	Upper Core Barrel
UCP	Upper Core Plate
USP	Upper Support Plate
UT	Ultrasonic Testing (a Volumetric NDE Method)
UTS	Upper Thermal Shield
VT	Visual Testing (a Visual NDE Method that Includes VT-1 and VT-3)
X	Existing, I&E Guidelines Component Group
XL	Extra-Long Westinghouse Fuel

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RECORD OF REVISION

0	Original Issue
A	This revision incorporates the Topical Report Conditions resulting from the NRC Safety Evaluation Review (see above) and responses to associated Requests for Additional Information (see Appendix B). It also contains minor editorial corrections identified since the original issuance of the guidelines. See Appendix C for detailed changes to the guidelines.

1
EXECUTIVE SUMMARY

Demonstration that the effects of aging degradation in pressurized water reactor (PWR) internals are adequately managed is essential for maintaining a healthy fleet and assuring continued functionality of the reactor internals. As a work product of the Electric Power Research Institute (EPRI) Materials Reliability Program (MRP) Reactor Internals Focus Group (RI-FG), these Inspection & Evaluation (I&E) guidelines are intended to support that demonstration, with requirements for inspection to detect the effects of aging degradation. These guidelines are provided to individual plant owners for use in preparing and executing their PWR internals aging management programs. These guidelines contain **Mandatory** and **Needed** requirements that must be implemented per the Materials Initiative [1]. Section 7 describes all of the requirements of the guidelines, including an implementation schedule. The requirements contained in this document are applicable to Babcock & Wilcox (B&W), Combustion Engineering (CE), and Westinghouse Nuclear Steam Supply System (NSSS) PWR designs currently operating in the United States.

These guidelines do not reduce, alter, or otherwise affect current American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Section XI [2] or plant-specific licensing inservice inspection requirements.

The program to develop these guidelines has been underway for almost a decade, organized around a framework and strategy for managing the effects of aging in PWR internals, dependent on a substantial database of material data and supporting evaluation results. The key sequential steps included the following:

- development of screening criteria, with susceptibility levels for the eight postulated aging mechanisms relevant to reactor internals and their effects;
- initial component screening and categorization, using the susceptibility levels to identify the relative susceptibility of the components;
- functionality assessment of degradation for components and assemblies of components;
- aging management strategy development combining the results of functionality assessment with component accessibility, operating experience, existing evaluations, and prior examination results to determine the appropriate aging management methodology, baseline examination timing, and the need for and the timing of subsequent inspections.

Through this process, the reactor internals for all three PWR designs were assigned to one of the following four groups: Primary, Expansion, Existing Programs, and No Additional Measures components. Definitions and recommendations for aging management actions specific to each group are provided in Sections 3 and 4.

The aging management elements needed for Primary and Expansion components were selected from existing, well-proven visual, surface, and volumetric examination methodologies that have been subject to widespread, relevant application. Each component in the Primary and Expansion groups was then assessed in terms of the degradation effect (e.g., cracking caused by particular mechanisms, loss of material caused by wear), appropriate examination methodology for detection of that effect, accessibility of that component for the examination method selected, and industry experience with those examinations. The Inspection Standard for PWR internals (MRP-228) [3] is the companion document to these I&E guidelines and provides the examination requirement standards for the components listed herein.

The Primary components requirements are listed in Tables 4-1, 4-2, and 4-3 of Section 4 for the Babcock & Wilcox (B&W), Combustion Engineering (CE), and Westinghouse designs, respectively. The Expansion components requirements are listed in Tables 4-4, 4-5, and 4-6 for the B&W, CE, and Westinghouse designs, respectively. These tables provide the assembly/sub-assembly/component description, the relevant degradation effect and associated degradation mechanism, any link between a Primary component and a related Expansion component, the examination method, and examination coverage.

The Existing Programs components requirements are listed in Tables 4-7, 4-8, and 4-9 for the B&W, CE, and Westinghouse designs, respectively. These tables and the supporting text identify the components and the references to the existing programs.

Tables are not provided for the No Additional Measures components. This group of components has been determined to need no additional aging management. However, for those components in the No Additional Measures group that are classified as core support structures in plant-specific documentation, the inservice inspection requirements of the ASME Code Section XI, Subsection IWB, Examination Category B-N-3 [2] must continue to be met, unless specific relief is granted as allowed by Title 10 Part 50.55a [4] of the Code of Federal Regulations (10CFR50.55a) or plant-specific licensing documentation.

The examination acceptance criteria and the expansion criteria for the primary/expansion links are described in Tables 5-1, 5-2, and 5-3 for the B&W, CE, and Westinghouse designs, respectively. These examination acceptance criteria include visual examination relevant conditions that require disposition by additional examinations, engineering evaluation, or repair/replacement.

Section 6 is for information and contains various options that are available for the disposition of conditions detected during examinations (Section 4) that are unable to satisfy the examination acceptance criteria (Section 5).

2
INTRODUCTION

2.1 Background

This document provides inspection and evaluation (I&E) guidelines for use by the industry to develop engineering programs to manage aging in Pressurized Water Reactor (PWR) internals. It is also intended to support the industry in developing an aging management program (AMP) for Pressurized Water Reactor (PWR) internals needed to satisfy license renewal commitments. Guidance for AMP preparation may be found in AMP XI.M16A of NUREG-1801, Revision 2 (or subsequent revisions).

The goal of these I&E guidelines is to ensure the long-term safety, integrity, and reliability of PWR internals using proven and familiar methods for inspection, monitoring, surveillance, and reporting. The guidelines are based on work performed over the past decade by the commercial nuclear power industry, first through the Joint Owners Baffle Bolt (JOBB) Program, then through the EPRI Materials Reliability Program (MRP) Reactor Internals Issue Task Group (RI-ITG) and, later, by the MRP Reactor Internals Focus Group (RI-FG). This program is organized around a framework and strategy [5] for managing the effects of aging in PWR internals, together with a substantial database of material data and supporting results (e.g., see [6]). The key steps in the framework and strategy process are shown in the flowchart of Figure 2-1.

Based upon the framework and strategy, and on the accumulated data, three important precursor elements to these I&E guidelines were then developed:

- screening criteria, considering chemical composition, neutron fluence exposure, temperature history, and representative stress levels, for determining the relative susceptibility of PWR internals to the eight postulated aging mechanisms [7] – stress corrosion cracking (SCC), irradiation-assisted stress corrosion cracking (IASCC), wear, fatigue, thermal aging embrittlement, irradiation embrittlement, irradiation-enhanced stress relaxation and creep, and void swelling;
- categorization of PWR internals, based on the screening criteria and the likelihood and severity of safety and economic consequences, into categories that range from those components for which these issues are insignificant (Category A) to those components that are potentially moderately significant (Category B) to those components that are potentially significantly affected (Category C) [8, 9, and 10]; and
- functionality assessment of components and assemblies of components based on representative plant designs using irradiated and aged material properties to determine the effects of the degradation mechanisms on functionality [11 and 12].

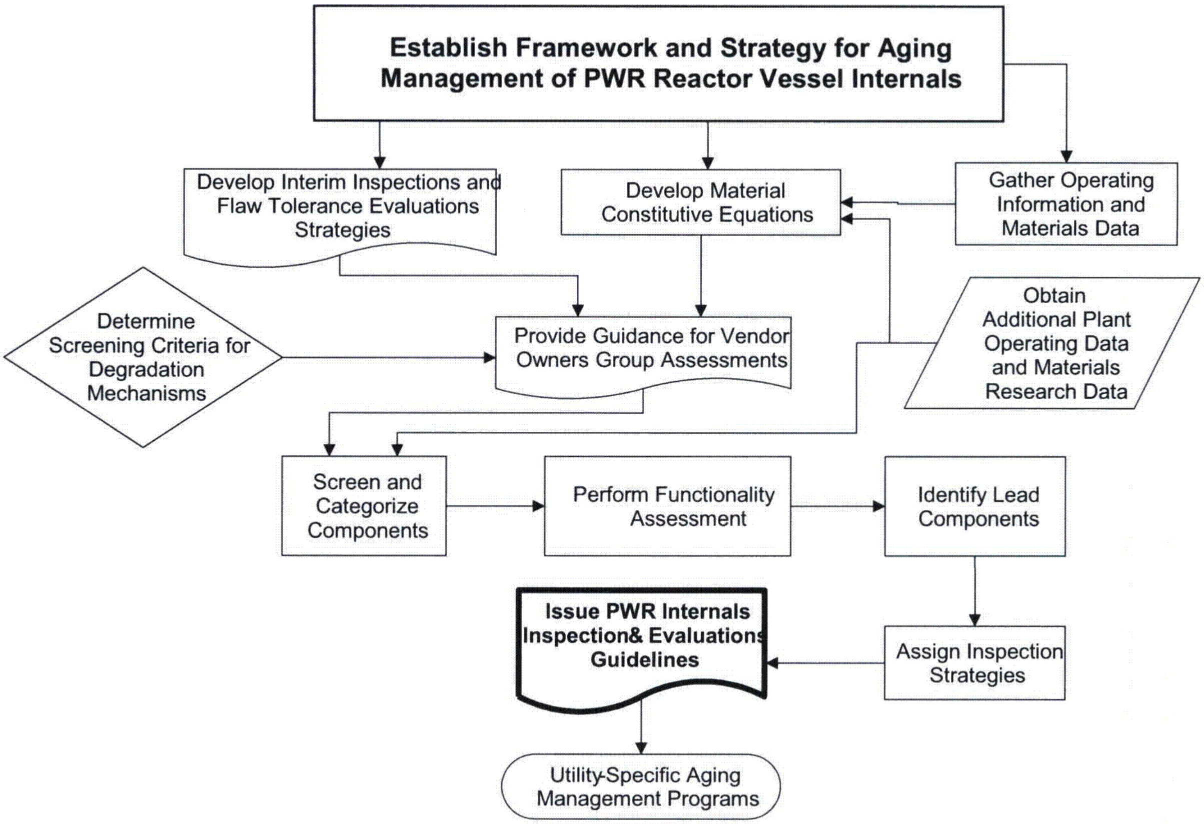

Figure 2-1
MRP framework and strategy for aging management of PWR internals

2.2 Aging Management Strategy Development

The aging management strategy development combined the results of functionality assessment with component accessibility, operating experience, existing evaluations, and prior examination results to determine the appropriate methodologies for maintaining the long-term functions of PWR internals safely and economically [13 and 14]. This process permitted further categorization of PWR internals into functional groups. Figure 2-2 shows the links between the categorization based on screening criteria, the functionality assessment, the aging management strategy development, and the I&E guidelines. The ultimate result of the process was to assign the components into Primary, Expansion, Existing Programs, and No Additional Measures groups, with appropriate recommendations to support AMP development. Complete definitions of these four groups are provided in Section 3.3.1.

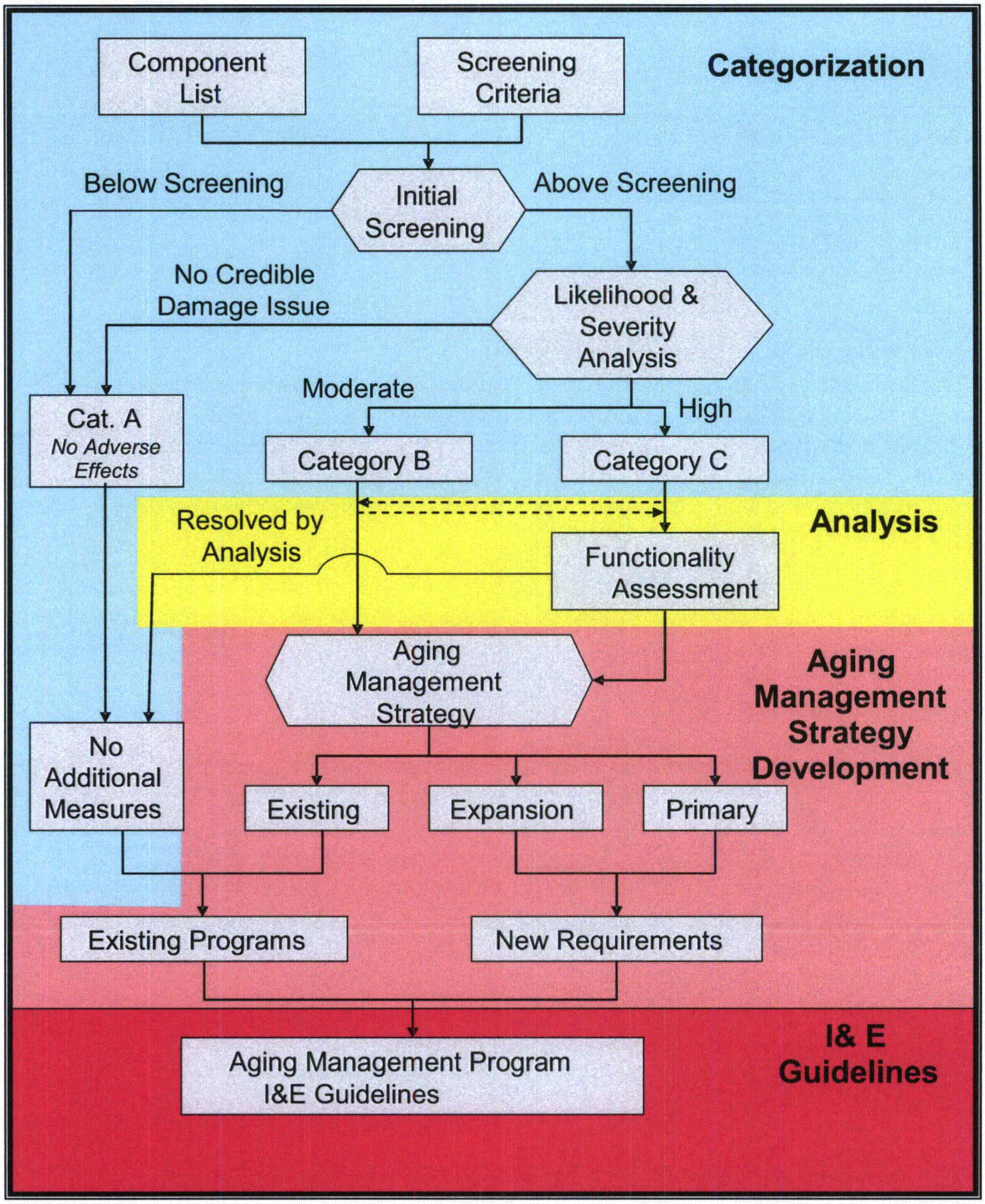

Figure 2-2
Links between categorization, functionality assessment, aging management strategy development and the I&E guidelines

2.3 Scope

These guidelines are intended to prescribe programs and activities that will assure the long-term safe and reliable operation of PWR internals as they age. As appropriately noted, the guidelines have requirements for both the original and the renewed licensing term (60-year plant life).

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. They are intended for operating commercial pressurized water reactors in the U.S., operated as base load generation units. These guidelines do not supersede or modify any plant-specific commitments without specific approval to do so by the regulatory body.

Section 3 provides a brief overview of currently licensed U.S. PWR internals – B&W, CE, and Westinghouse – that further defines the scope of these I&E guidelines. Section 4 identifies the components and inspection requirements. The examination acceptance criteria and the expansion criteria for the primary/expansion links are described in Section 5. Section 6 is for information and contains various options that are available for the disposition of conditions detected during examinations (Section 4) that are unable to satisfy the examination acceptance criteria (Section 5).

The implementation of these guidelines is governed by the Materials Guidelines Implementation Protocol (Appendix B) of NEI 03-08 [1]. The Mandatory and Needed requirements are summarized in Section 7.

2.4 Guidelines Applicability

The guidelines are intended to serve as the primary basis for owner preparation of an engineering program for managing aging in reactor internal components in accordance with the requirement cited in Section 7. The guidelines also serve as the primary basis for preparation of an Aging Management Program (AMP) for reactor internal components to support license renewal. 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.

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 assessments 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.