EPRI | ELECTRIC POWER RESEARCH INSTITUTE

MRP Materials Reliability Program MRP 2011-036

January 9, 2012

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

Attention: Sheldon Stuchell

Subject: **TRANSMITTAL: PWR REACTOR INTERNALS INSPECTION AND EVALUATION GUIDELINES (MRP-227-A)**

REFERENCE: EPRI PROJECT NUMBER 0669

1. *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-Rev. 0).* EPRI, Palo Alto, CA: 2008. 1016596.
2. U. S. Nuclear Regulatory Commission letter "Revision 1 to the Final Safety Evaluation of EPRI Report, Material Reliability Program Report 1016596 (MRP-227), Revision 0, *Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines*, (TAC NO. ME0680)," dated December 16, 2011.

Enclosed is the EPRI Technical Report "Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)." This non-proprietary report is forwarded for confirmatory review. MRP-227-A is a revision to the original version of the guidelines (MRP-227 Revision 0, Reference 1) that incorporates the Topical Report Conditions resulting from the U. S. Nuclear Regulatory Commission Safety Evaluation Review (Reference 2) and responses to associated Requests for Additional Information (see Appendix B of MRP-227-A).

This document contains *Mandatory* and *Needed* elements as defined in the Implementation Protocol of NEI 03-08 and is required for all operating commercial U.S. pressurized water reactors (PWRs). The *Mandatory* and *Needed* elements are described in more detail below.

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 and apply to both the current license period as well as to plants that have been granted license extensions through the license renewal process. 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 or plant-specific licensing in-service inspection requirements.

MRP-227-A contains one *Mandatory* element described as follows:

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

MRP-227-A contains five *Needed* elements described as follows:

- **Needed:** *Each commercial U.S. PWR unit shall implement Tables 4-1 through 4-9 and Tables 5-1 through 5-3 of MRP-227-A for the applicable design within twenty-four months following issuance of MRP-227-A (that is, no later than January 3, 2014).*
- **Needed:** *Examinations specified in these guidelines shall be conducted in accordance with the Inspection Standard (MRP-228).*
- **Needed:** *Examination results that do not meet the examination acceptance criteria defined in Section 5 of MRP-227-A shall be recorded and entered in the plant corrective action program and dispositioned.*
- **Needed:** *Each commercial U.S. PWR unit shall provide a summary report of all inspections and monitoring, items requiring evaluation, and new repairs to the MRP Program Manager within 120 days of the completion of an outage during which PWR internals within the scope of MRP-227 are examined.*
- **Needed:** *If an engineering evaluation is used to disposition an examination result that does not meet the examination acceptance criteria in Section 5 of MRP-227-A, this engineering evaluation shall be conducted in accordance with a NRC-approved evaluation methodology.*

The effective date of the requirements is January 3, 2012.

If you have any questions on this transmittal or on MRP-227-A, please contact Rick Reid by phone at (704) 595-2770 or by e-mail at rreid@epri.com.

Sincerely,

Scot A. Greenlee Digitally signed by Scot A. Greenlee
DN: cn=Scot A. Greenlee, o=Exelon, ou=Nuclear Generation, email=scot.greenlee@exeloncorp.com, c=US
Date: 2012.01.07 14:00:34 -06'00'

Scot Greenlee
Exelon Nuclear Corporation
Materials Reliability Program Executive Sponsor

cc: (next page)

cc: Jim Lash, FirstEnergy Nuclear Operating Company
Matt Sunseri, Wolf Creek Nuclear Operating Company
Tim Wells, Southern Company
Victoria Anderson, NEI
Randall Crane, INPO
Kurt Edsinger, EPRI
Anne Demma, EPRI
Robin Dyle, EPRI

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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)

1022863

Final Report, December 2011

EPRI Project Managers
A. Demma
R. Reid

This document does **NOT** meet the requirements of
10CFR50 Appendix B, 10CFR Part 21,
ANSI N45.2-1977 and/or the intent of ISO-9001 (1994)

ELECTRIC POWER RESEARCH INSTITUTE
3420 Hillview Avenue, Palo Alto, California 94304-1338 • PO Box 10412, Palo Alto, California 94303-0813 • USA
800.313.3774 • 650.855.2121 • askepri@epri.com • www.epri.com

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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

cc w/encl: See next page

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
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Office of Nuclear Reactor Regulation

Project No. 669

Enclosure:
Final SE

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ADAMS ACCESSION NO.: ML11308A770 **TAC ME7328**

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NAME	SStuchell	DBaxley CMH for	MMitchell	JJolicoeur	RNelson
DATE	11 /16/11	12/15/11	11/16/11	12/15/11	12/16/11

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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.

Principal Contributor: Ganesh Cheruvenki

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CITATIONS

This report was prepared by

MRP Reactor Internals Inspection and Evaluation Guidelines Core Group

Tim Wells, *Chairman*	Southern Nuclear Operating Co.
Cheryl Boggess	Westinghouse Electric Company, LLC
Bernie Rudell	Constellation Energy
Anne Demma, *Program Manager*	EPRI
Rick Reid, *Project Manager*	EPRI
Steve Fyfitch	AREVA NP Inc.
Steve Herman	AREVA NP Inc.
Glenn Gardner	Dominion Nuclear Connecticut
Mark Joseph	Florida Power & Light Co.
Randy Lott	Westinghouse Electric Company, LLC
Robert Nickell, *Principal Investigator*	Anatech Corp.
Jack Spanner	EPRI
Chuck Welty	EPRI
David Whitaker	Duke Energy
Hongqing Xu	AREVA NP Inc.

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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].

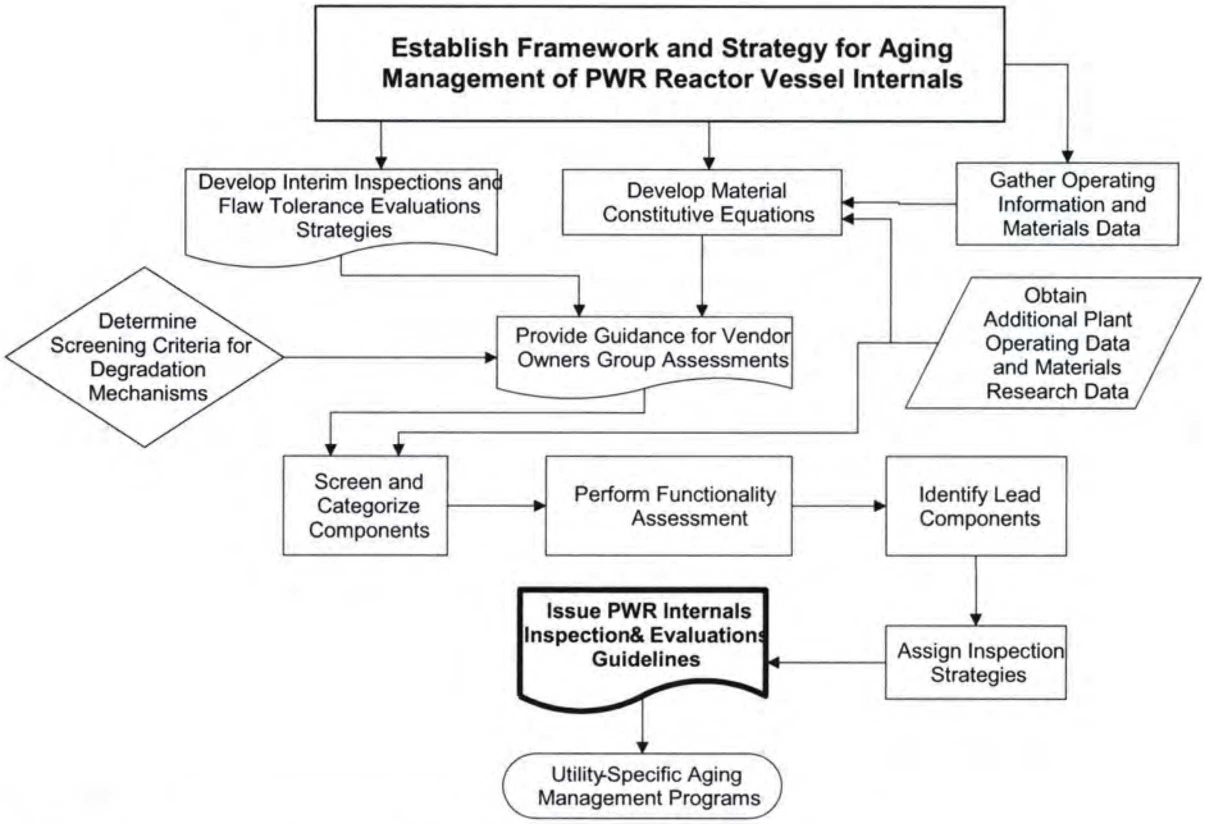

Figure 2-1
MRP framework and strategy for aging management of PWR internals

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.

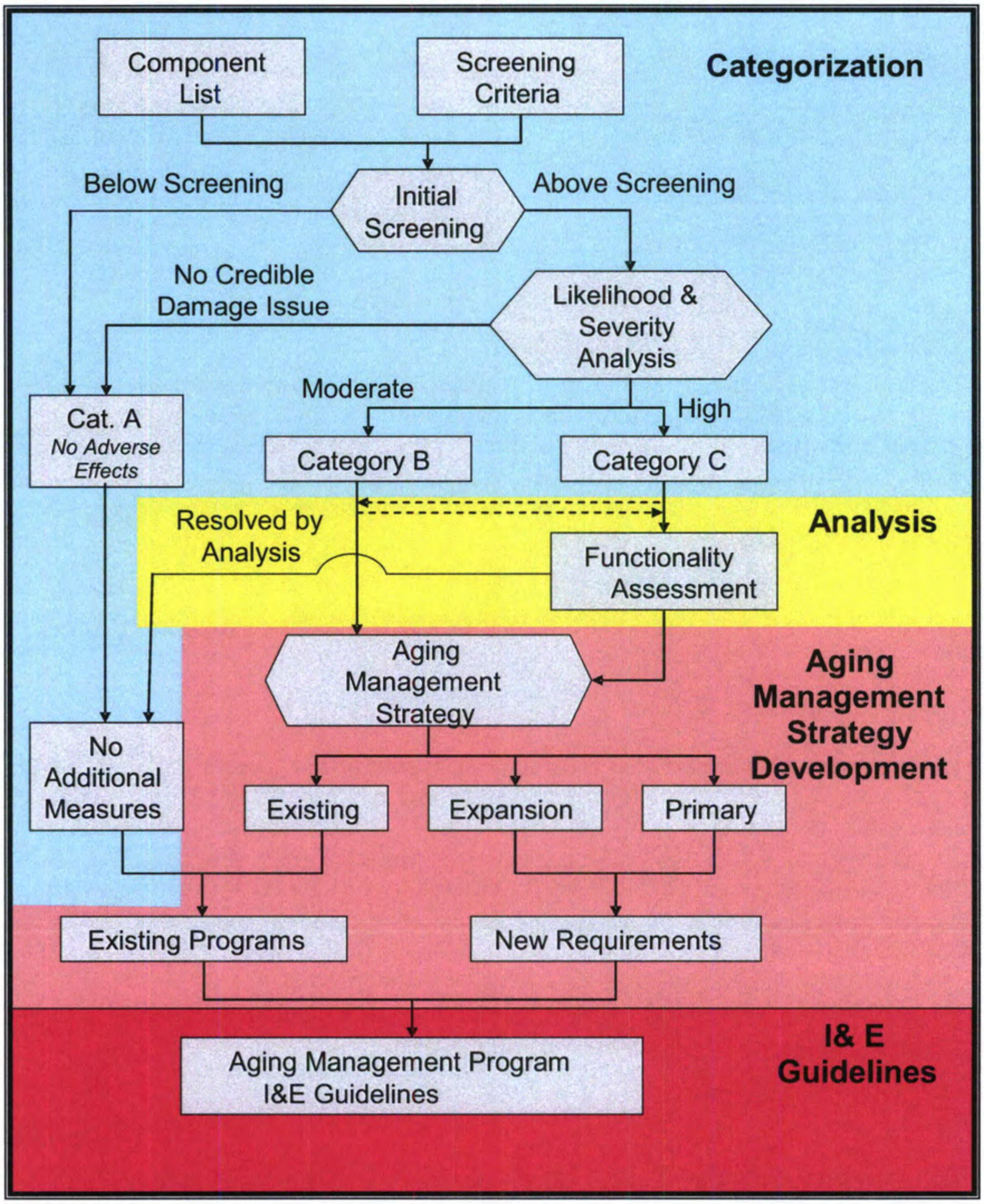

Figure 2-2
Links between categorization, functionality assessment, aging management strategy development and the I&E guidelines

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.

3
COMPONENT CATEGORIZATION AND AGING MANAGEMENT STRATEGY DEVELOPMENT

This section of the I&E guidelines provides a summary of the design characteristics for B&W, CE, and Westinghouse PWR internals; a summary of the screening process used for the preliminary categorization of PWR internals; and a summary of the categorization and aging management strategy development results.

3.1 Design Characteristics Summary

The functions of PWR internals are to:

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

3.1.1 B&W Internals Design Characteristics

The seven B&W-designed operating units share common design characteristics with minor variations. The B&W-designed PWR internals consist of two major structural assemblies that are located within, but not welded to the reactor vessel. These two major assemblies are called the plenum assembly and the core support assembly (CSA). The latter includes three principal sub-assemblies – the core support shield (CSS) assembly, the core barrel assembly, and the lower internals assembly. The general arrangement of the B&W-designed PWR internals is shown in Figure 3-1. A brief summary of the design characteristics for these internals is provided in the following sub-section. For a more complete discussion, see Reference 8.

Plenum Assembly

The plenum assembly is a cylindrical structure with perforated grid plates on top and bottom, and is comprised of: (1) the plenum cover assembly; (2) the plenum cylinder assembly; (3) the upper grid assembly; and (4) the control rod guide tube assemblies. The plenum assembly fits inside the core support shield, positions the top of the fuel assemblies, supports the control rod guide tube assemblies, and provides the core hold-down required for hydraulic lift forces. The plenum assembly also provides continuous guidance and protection for the control rods, and directs flow out of the core to reactor vessel outlet nozzles. The plenum assembly is removed at the beginning of every refueling outage, in order to permit access to the fuel assemblies.

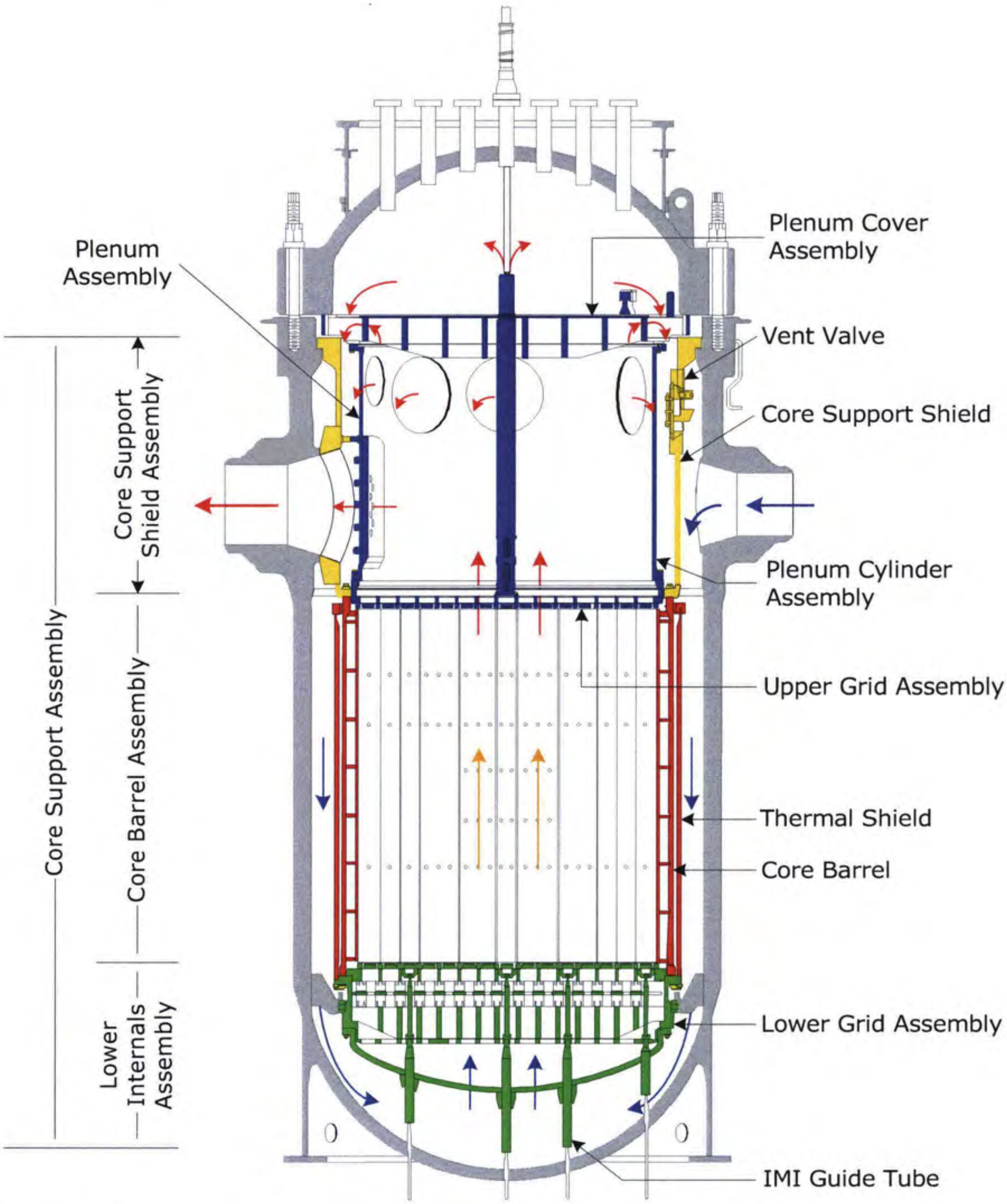

Figure 3-1
Overview of typical B&W internals

The plenum cover assembly is bolted to the top of the plenum cylinder, and consists of a weldment, a bottom flange, a support ring and flange, a cover plate, and lifting lugs. The plenum cover assembly provides support for the top of the control rod guide tube assemblies. The lifting lugs are used to lift the plenum assembly out of the reactor vessel.

The plenum cylinder assembly is bolted to the bottom of the plenum cover assembly and consists of a cylinder, top and bottom flanges, reinforcing plates, and round bars. Its function is to direct the flow of reactor coolant from the core region to the reactor vessel outlet nozzles.

The upper grid assembly sits inside the lower flange of the core support shield and is bolted to the plenum cylinder bottom flange. It is comprised of an upper grid ring forging, an upper grid rib section, and fuel assembly support pads. Its function is to support and provide a seating surface for the tops of the fuel assemblies located within the core barrel below, and to restrain and align the bottoms of the control rod guide tubes.

The control rod guide tube assemblies each consist of a pipe (the guide housing), a flange, spacer castings, guide tubes, and rod guide sectors. The assemblies are welded to the plenum cover plate and bolted to the upper grid assembly. Their function is to provide control rod assembly guidance, protect the control rod assembly from the effects of potential coolant cross-flow, and structurally connect the upper grid assembly to the plenum cover.

Core Support Assembly

The core support assembly is fabricated by bolting together the core support shield assembly, the core barrel assembly, and the lower internals assembly to form a tall cylinder. The core support assembly remains in place in the reactor vessel during refueling, and is removed only to perform scheduled inspections of the reactor vessel interior surfaces or of the core support assembly itself.

The top portion of the core support assembly is the core support shield assembly, a cylinder with an upper flange that rests on a circumferential support ledge in the reactor vessel closure flange, thereby supporting the entire core support assembly. It sits directly on top of the core barrel, and consists of a cylinder, top and bottom flanges, outlet nozzles, vent valve nozzles, vent valves, round bars, flow deflectors, and lifting lugs. Its function is to provide a boundary between the incoming cold reactor coolant on the outside of the cylinder and the heated reactor coolant flowing on the inside of the cylinder.

The core barrel assembly is a second flanged cylinder, with its top flange bolted to the bottom flange of the core support shield assembly and its bottom flange bolted to the top flange of the lower internals assembly. The core barrel assembly consists of a cylinder, top and bottom flanges, baffle and former plates, and a thermal shield cylinder. Its functions are to direct the flow of coolant and to support the lower internals assembly. In addition, the thermal shield reduces the amount of radiation that reaches the reactor vessel. The incoming reactor coolant is directed downward along the outside of the core barrel cylinder and upward through the fuel assemblies contained inside the core barrel. A small amount of coolant flows upward through the space between the core barrel cylinder and the baffle plates. A small portion of the coolant also runs down the annulus between the thermal shield and the core barrel cylinder, through holes drilled in the core barrel cylinder bottom flange, and then upward through the core.

The lower internals assembly consists of a lower grid assembly, a flow distributor assembly, and in-core monitoring instrumentation guide tube assemblies. The lower internals assembly is bolted to the bottom flange of the core barrel cylinder, and its function is to direct coolant flow upward through the fuel assemblies. The lower grid assembly consists of three grid structures or flow plates: (1) the lower grid rib section, (2) the flow distributor plate, and (3) the lower grid forging. Each of these flow plates has holes or flow ports to direct coolant flow upward toward the fuel assemblies.

3.1.2 CE Internals Design Characteristics

In general, the 14 operating CE-designed PWRs in the U.S. are divided into three groups: (1) those with a bolted core shroud and top-mounted in-core instrumentation (ICI); (2) those with a welded core shroud and top-mounted ICI; and (3) those with a welded core shroud and bottom-mounted ICI.

The CE-designed PWR internals consist of three major structural assemblies, plus three other sets of major components. The three major assemblies are the: (1) upper internals assembly, (2) core support barrel assembly, and (3) lower internals assembly. In addition, the three other sets of major components are the control element assembly shroud assemblies, core shroud assembly, and in-core instrumentation support system. The general arrangement of the CE-designed PWR internals is shown in Figure 3-2. A brief summary of the design characteristics for these internals is provided in the following sub-section. For a more complete discussion, see Reference 10.

Upper Internals Assembly

The upper internals assembly is located above the reactor core, within the core support barrel assembly, and is removed during refueling as a single component in order to provide access to the fuel assemblies. The upper internals assembly consists of the upper guide structure support plate, the fuel assembly alignment plate, the control element assembly shroud assemblies, the upper guide structure grid assembly, the upper guide structure cylinder, the in-core instrumentation support system and the hold-down ring (or expansion compensating ring). The functions of the upper internals assembly are to provide alignment and support to the fuel assemblies, to maintain control element assembly shroud spacing, to prevent movement of the fuel assemblies in the case of a severe accident condition, and to protect the control rods from cross-flow effects in the upper plenum. The flange on the upper end of the upper internals assembly rests on the core support barrel.

Core Support Barrel

The core support barrel assembly consists of the core support barrel, the core support barrel upper flange, core support barrel alignment keys, and the core support barrel snubbers. In one CE plant, a thermal shield is part of the core support barrel assembly.

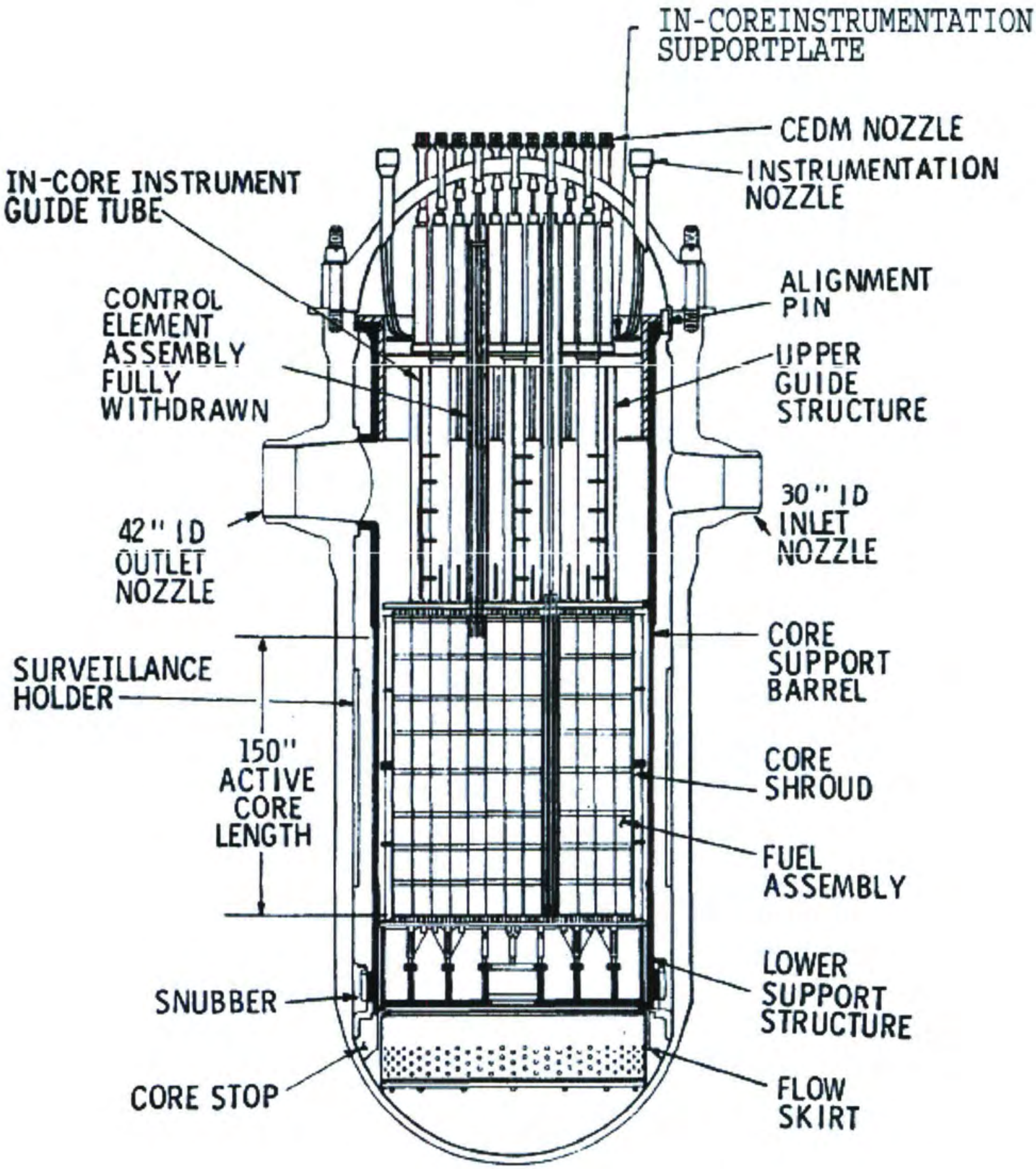

Figure 3-2
Overview of typical CE internals

The core support barrel is a cylinder which contains the core and other internals. Its function is to resist static loads from the fuel assemblies and other internals, and dynamic loads from normal operating hydraulic flow, seismic events, and loss-of-coolant-accident (LOCA) events. The core support barrel also supports the lower internals assembly and its core support plate, upon which the fuel assemblies rest.

The core support barrel upper flange is a thick ring that supports and suspends the core support barrel from a ledge on the reactor vessel.

Lower Internals Assembly

The lower internals assembly consists of the core support plate, the fuel alignment pins, the core support columns, the in-core instrumentation (ICI) support system, and the lower support structure beam assemblies. The core support plate functions are to position and support the reactor core, and to provide control of reactor coolant flow into each fuel assembly. The core support plate transmits the weight of the core to the core support barrel by means of the vertical core support columns, an annular skirt, and the lower support structure beams. The fuel alignment pins protrude from the core support plate and provide guidance and limit lateral movement of the individual fuel assemblies. CE plants with a welded core shroud and bottom-mounted ICI have no core support plate, in which case the fuel alignment pins are attached directly to the core support deep beams.

Core Shroud Assembly

The core shroud assembly is located within the core support barrel and directly below the upper internals assembly. The core shroud assembly is attached to the core support barrel by threaded structural fasteners for those internals with a bolted core shroud and top-mounted ICI. The core shroud assembly is attached to the core support plate – an element of the lower internals assembly – by tie rods or welds for the internals with a welded core shroud and top-mounted ICI (Figure 3-3). The core shroud assembly is attached to the lower internals assembly cylinder by welding for those internals with a welded core shroud and bottom-mounted ICI (Figure 3-4). The core shroud assembly functions are to provide a boundary between reactor coolant flow on the outside of the core support barrel and the reactor coolant flow through the fuel assemblies, to limit the amount of coolant bypass flow, and to reduce the lateral motion of the fuel assemblies.

Control Element Assembly Shroud Assemblies

The control element assembly shroud assemblies consist of control element assembly shrouds, the control element assembly shroud bolts, and the control element assembly shroud extension shaft guides. The shroud tubes protect the control rods from cross-flow effects in the upper plenum. The bottom part of the shrouds is bolted at their lower end to the fuel assembly alignment plate. The extension shaft guides also protect the control rods from cross-flow effects in the upper plenum, and provide lateral support and alignment of the control element assembly extension shafts during refueling operations. The control element drive mechanisms are positioned on the reactor vessel closure head and are coupled to the control element assemblies by the control element assembly extension shafts. Control element assembly shroud assemblies are attached to the upper guide structure support plate by tie rods.

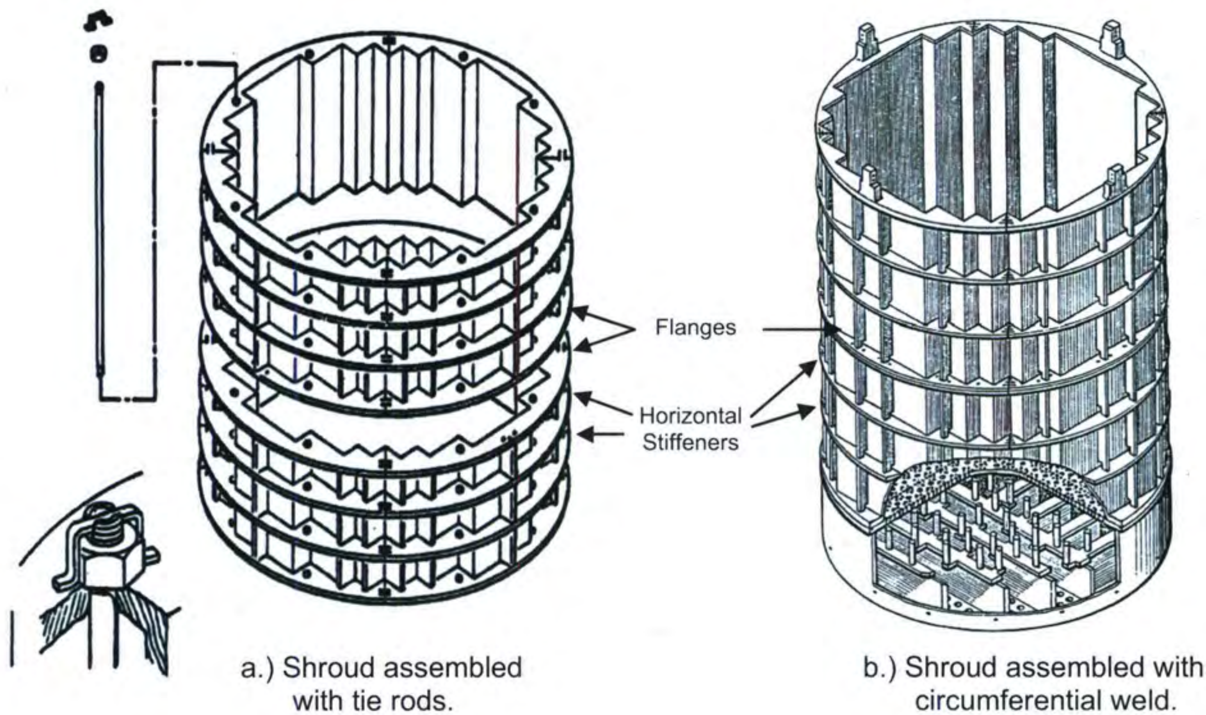

Figure 3-3
CE welded core shroud designs assembled in two vertical sections (with top-mounted ICI)

In-Core Instrumentation Support System

The in-core instrumentation support system consists of in-core instrumentation guide tubes and components which provide support to the in-core instrumentation.

For plants with top-entry in-core instrumentation assemblies, the in-core instrumentation is inserted through the reactor vessel head through a nozzle into a guide tube. The guide tubes interface with the thimble support plate, which is perforated to fit over the control element assembly extension shaft guides, with a connection to the upper guide structure support plate. ICI thimble tube assemblies extend downward from a flanged connection at the thimble support plate (in the original design) through the fuel alignment plate and into the reactor core. The upper portion of the ICI thimble tube exists between the thimble support plate and fuel alignment plate, while the lower ICI thimble tube is the zirconium alloy portion that extends into the fuel assemblies.

For plants with bottom-entry in-core instrumentation, the guide tubes are connected to and supported by the lower internals assembly, from which the in-core instrumentation enters the core.

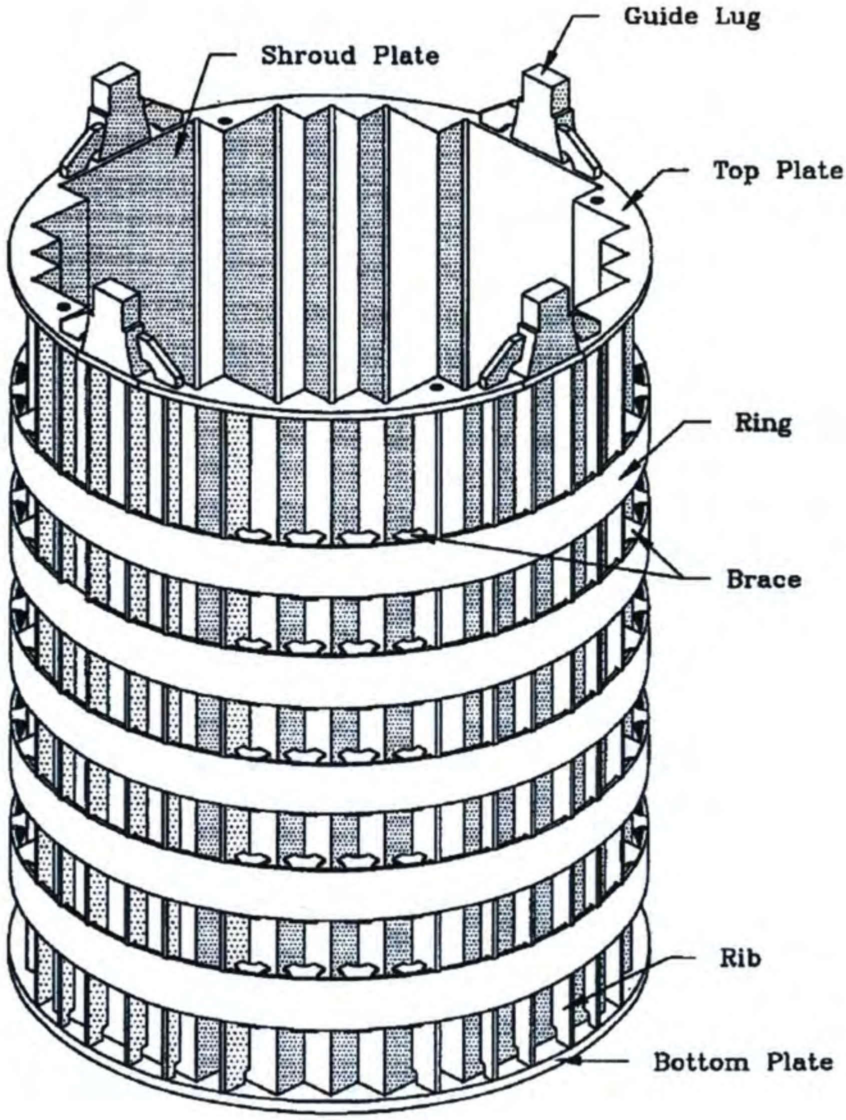

Figure 3-4
CE welded core shroud with full height panels (with bottom-mounted ICI)

3.1.3 Westinghouse Internals Design Characteristics

A schematic view of a typical set of Westinghouse-designed PWR internals is shown in Figure 3-5. However, because of the significant variation in design characteristics, the 48 operating Westinghouse PWRs in the U.S. are sub-divided into various groups, starting with the number of reactor coolant system (RCS) loops – two-loop, three-loop, and four-loop configurations. Other significant variations include the original thermal output, the baffle-barrel region flow design (downflow, upflow, and converted upflow), and upper support plate configuration. A complete set of these groups is provided in Section 4 of Reference 10.

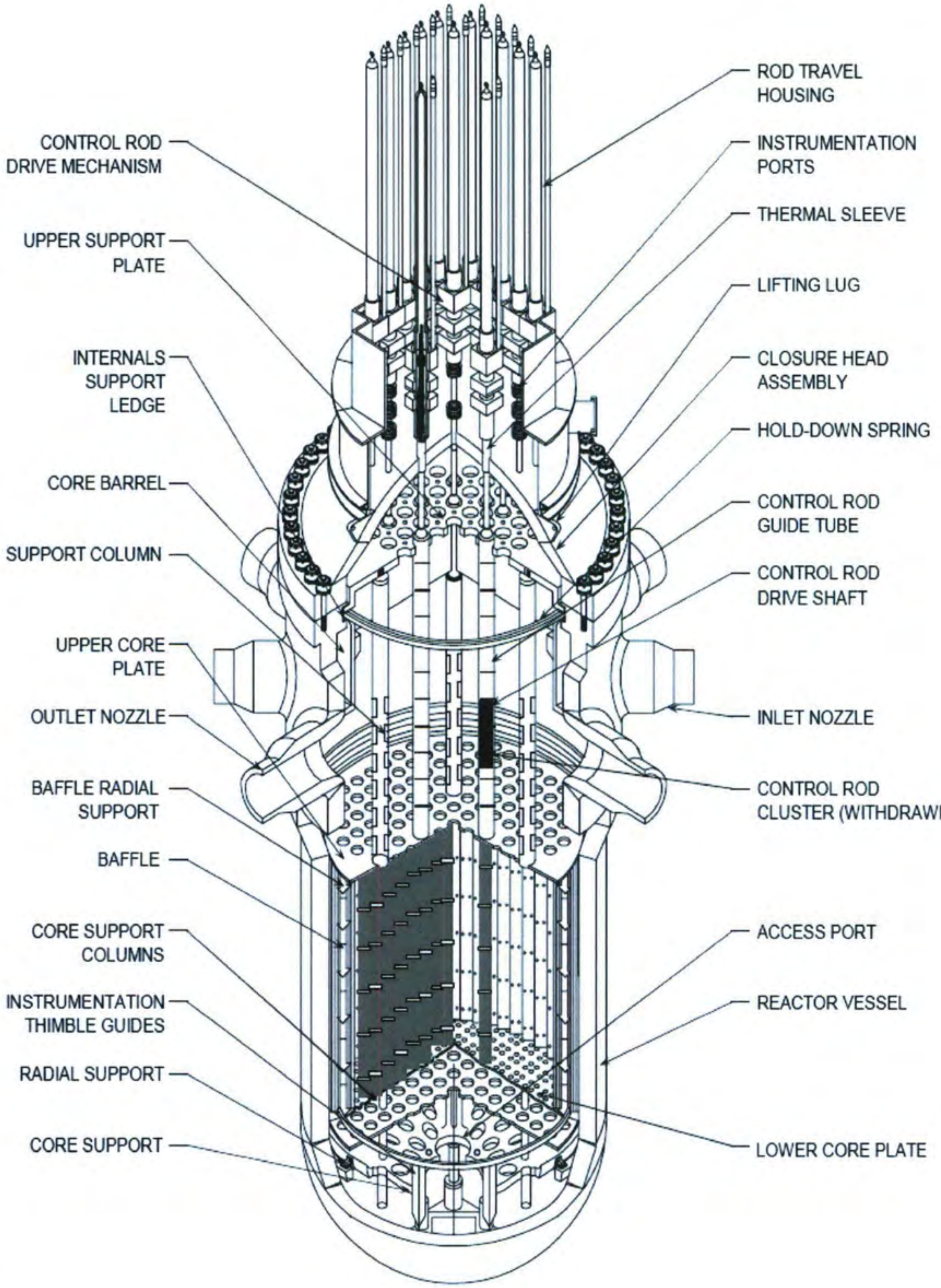

Figure 3-5
Overview of typical Westinghouse internals

All Westinghouse internals consist of two basic assemblies: an upper internals assembly that is removed during each refueling operation to obtain access to the reactor core and a lower internals assembly that can be removed following a complete core off-load.

The lower internals assembly is supported in the vessel by clamping to a ledge below the vessel-head mating surface and is closely guided at the bottom by radial support/clevis assemblies. The upper internals assembly is clamped at this same ledge by the reactor vessel head. The bottom of the upper internals assembly is closely guided by the core barrel alignment pins of the lower internals assembly.

Upper Internals Assembly

The major sub-assemblies that comprise the upper internals assembly are the: (1) upper core plate (UCP) and fuel alignment pins; (2) upper support column assemblies; (3) control rod guide tube assemblies and flow downcomers; (4) upper plenum; and (5) upper support plate assembly.

During reactor operation, the upper internals assembly is preloaded against the fuel assembly springs and the internals holddown springs by the reactor vessel head pressing down on the outside edge of the upper support plate (USP). The USP acts as the divider between the upper plenum and the reactor vessel head and as a relatively stiff base for the rest of the upper internals. The upper support columns and the guide tubes are attached to the USP. The UCP, in turn, is attached to the upper support columns. The USP assemblies are designated as one of three different designs: (1) a deep beam design, (2) a top hat design, or (3) an inverted top hat design.

The UCP is perforated to permit coolant to pass from the core below into the upper plenum defined by the USP and the UCP. The coolant then exits through the outlet nozzles in the core barrel. The UCP positions and laterally supports the core by fuel alignment pins extending below the plate. The UCP contacts and preloads the fuel assembly springs and thus maintains contact of the fuel assemblies with the lower core plate (LCP) during reactor operation.

The upper support columns vertically position the UCP and are designed to take the uplifting hydraulic flow loads and fuel spring loads on the UCP. The guide tubes are bolted to the USP and pinned at the UCP so they can be easily removed if replacement is desired. The guide tubes are designed to guide the control rods in and out of the fuel assemblies to control power generation. The guide tubes are also slotted in their lower sections to allow coolant exiting from the core to flow into the upper plenum.

The upper instrumentation columns are bolted to the USP. These columns support the thermocouple guide tubes that lead the thermocouples from the reactor head into the upper plenum to just above the UCP.

The UCP alignment pins locate the UCP laterally with respect to the lower internals assembly. The pins must laterally support the UCP so that the plate is free to expand radially and move axially during differential thermal expansions between the upper internals and the core barrel. The UCP alignment pins are the interfacing components between the UCP and the core barrel. The UCP alignment pins are shrunk-fit and welded into the core barrel and the core barrel bearing pad. The gap sizes between the alignment pins and the matching inserts are customized.

The USP, the upper support columns, and the UCP are typically considered core support structures.

Lower Internals Assembly

The reactor core is positioned and supported by the lower internals and upper internals assemblies. The individual fuel assemblies are positioned by fuel alignment pins in the LCP and in the UCP. These pins control the orientation of the core with respect to the lower internals and upper internals assemblies. The lower internals are aligned with the upper internals by the UCP alignment pins and secondarily by the head/vessel alignment pins. The lower internals are orientated to the vessel by the lower radial keys and by the head/vessel alignment pins. Thus, the core is aligned with the vessel by a number of interfacing components.

The lower internals assembly is supported in the vessel by clamping to a ledge below the vessel-head mating surface and closely guided at the bottom by radial support/clevis assemblies. The upper internals assembly is clamped at this same ledge by the reactor vessel head. The bottom of the upper internals assembly is closely guided by the core barrel alignment pins of the lower internals assembly.

The fuel assemblies are supported inside the lower internals assembly on top of the LCP. The LCP is elevated above the lower support forging by support columns and bolted to a ring support attached to the inside diameter of the core barrel. The support columns transmit vertical fuel assembly loads from the LCP to the much thicker lower support forging. The lower support forging is welded to and supported by the core barrel, which transmits vertical loads to the vessel through the core barrel flange.

The functions of the LCP are to position and support the core and provide a metered control of reactor coolant flow into each fuel assembly. The LCP is located near the bottom of the lower support assembly, inside the core barrel, and above the lower support forging.

The function of the lower support forging or casting is to provide support for the core. The lower support forging is attached with a full-penetration weld to the lower end of the core barrel. In this position it can provide uninterrupted support to the core. The core sits directly on the LCP, which is supported by the lower support columns that are attached to and extend above the lower support forging. Some four-loop plants employ a cast lower support instead of a forging. The functions, loads, and supporting hardware are the same except for dimensions.

The primary function of the core barrel is to support the core. A large number of components are attached to either the core barrel or the core barrel flange, including the baffle/former assembly, the outlet nozzles, the neutron panel assemblies or thermal shield, the alignment pins that engage the UCP and the LCP, the lower support forging, and the LCP. The radial keys restrain large transverse motions of the core barrel but at the same time allow unrestricted radial and axial thermal expansions.

The baffle and former assembly is made up of vertical plates called baffles and horizontal support plates called formers. The baffle plates are bolted to the formers by the baffle/former bolts, and the formers are attached to the core barrel inside diameter by the barrel/former bolts. The baffle/former assembly forms the interface between the core and the core barrel. The baffles provide a barrier between the core and the former region so that a high concentration of flow in the core region can be maintained. A secondary benefit, although not a requirement of the baffles, is to reduce the neutron flux on the vessel.

Baffle plates are secured to each other at selected corners by edge bolts. In addition, in some installations, corner brackets are installed behind and bolted to the baffle plates.

The function of the core barrel outlet nozzles is to direct the reactor coolant, after it leaves the core, radially outward through the reactor vessel outlet nozzles. The core barrel outlet nozzles are located in the upper portion of the core barrel directly below the flange and are attached to the core barrel, each in line with a vessel outlet nozzle.

Additional neutron shielding of the reactor vessel is provided in the active core region by neutron panels or thermal shields that are attached to the outside of the core barrel. Specimen guides that contain specimens for determining the irradiation effects of the vessel during the life of the plant are attached to the neutron panels/thermal shields.

The flux thimble is a long, slender stainless steel tube that passes from an external seal table, through the bottom mounted nozzle penetration, through the lower internals assembly, and finally extends to the top of the fuel assembly. The flux thimble provides a path for the neutron flux detector into the core and is subjected to reactor coolant pressure and temperature on the outside surface and to atmospheric conditions on the inside. The flux thimble path from the seal table to the bottom mounted nozzles is defined by flux thimble guide tubes, which are part of the primary pressure boundary and not considered to be part of the internals. The bottom-mounted instrumentation (BMI) columns provide a path for the flux thimbles from the bottom of the vessel into the core. The BMI columns align the flux thimble path with instrumentation thimbles in the fuel assembly.

The LCP and the fuel alignment pins, the lower support forging or casting, the lower support columns, the core barrel, the core barrel flange, the radial support keys, the baffle plates, and the former plates are typically classified as core support structures.

3.2 Initial Screening Summary

This sub-section contains a summary of the initial screening of PWR internals – screening those internals on the basis of susceptibility to eight different age-related degradation mechanisms – stress corrosion cracking (SCC), irradiation-assisted stress corrosion cracking (IASCC), wear, fatigue, thermal aging embrittlement, irradiation embrittlement, void swelling, and the combination of thermal and irradiation-enhanced stress relaxation or irradiation-enhanced creep. Development and justification of the screening criteria required knowledge of the specific aging mechanisms and their effects, some engineering judgment, and the use of empirical relations where data were lacking. The full explanation of the screening criteria for the eight age-related degradation mechanisms identified for PWR internals is provided in Reference 7.

For this initial screening, the group of PWR internals that were deemed not to be susceptible to any of the eight age-related degradation mechanisms (i.e., below the screening criteria) were placed into the A Category. The Category A components are listed in previous reports for the B&W PWR designs [8] and the CE and Westinghouse PWR designs [10]. The further categorization of the components is discussed in Section 3.3.

The age-related degradation mechanisms used for the initial screening are defined in the following sub-sections. More detailed discussions of these aging mechanisms are provided in Reference 7.

3.2.1 Stress Corrosion Cracking

Stress Corrosion Cracking (SCC) refers to local, non-ductile cracking of a material due to a combination of tensile stress, environment, and metallurgical properties. The actual mechanism that causes SCC involves a complex interaction of environmental and metallurgical factors. The aging effect is cracking.

3.2.2 Irradiation-Assisted Stress Corrosion Cracking

Irradiation-assisted stress corrosion cracking (IASCC) is a unique form of SCC that occurs only in highly-irradiated components. The aging effect is cracking.

3.2.3 Wear

Wear is caused by the relative motion between adjacent surfaces, with the extent determined by the relative properties of the adjacent materials and their surface condition. The aging effect is loss of material.

3.2.4 Fatigue

Fatigue is defined as the structural deterioration that can occur as the result of repeated stress/strain cycles caused by fluctuating loads and temperatures. After repeated cyclic loading of sufficient magnitude, microstructural damage can accumulate, leading to macroscopic crack initiation at the most highly affected locations. Subsequent mechanical or thermal cyclic loading can lead to growth of the initiated crack. Corrosion fatigue is included in the degradation description.

Low-cycle fatigue is defined as cyclic loads that cause significant plastic strain in the highly stressed regions, where the number of applied cycles is increased to the point where the crack eventually initiates. When the cyclic loads are such that significant plastic deformation does not occur in the highly stressed regions, but the loads are of such increased frequency that a fatigue crack eventually initiates, the damage accumulated is said to have been caused by high-cycle fatigue. The aging effects of low-cycle fatigue and high-cycle fatigue are additive.

Fatigue crack initiation and growth resistance is governed by a number of material, structural and environmental factors, such as stress range, loading frequency, surface condition and presence of deleterious chemical species. Cracks typically initiate at local geometric stress concentrations, such as notches, surface defects, and structural discontinuities. The aging effect is cracking.

3.2.5 Thermal Aging Embrittlement

Thermal aging embrittlement is the exposure of delta ferrite within cast austenitic stainless steel (CASS) and precipitation-hardenable (PH) stainless steel to high inservice temperatures, which can result in an increase in tensile strength, a decrease in ductility, and a loss of fracture toughness. Some degree of thermal aging embrittlement can also occur at normal operating temperatures for CASS and PH stainless steel internals. CASS components have a duplex microstructure and are particularly susceptible to this mechanism. While the initial aging effect is loss of ductility and toughness, unstable crack extension is the eventual aging effect if a crack is present and the local applied stress intensity exceeds the reduced fracture toughness.

3.2.6 Irradiation Embrittlement

Irradiation embrittlement is also referred to as neutron embrittlement. When exposed to high-energy neutrons, the mechanical properties of stainless steel and nickel-base alloys can be changed. Such changes in mechanical properties include increasing yield strength, increasing ultimate strength, decreasing ductility, and a loss of fracture toughness. The irradiation embrittlement aging mechanism is a function of both temperature and neutron fluence. While the initial aging effect is loss of ductility and toughness, unstable crack extension is the eventual aging effect if a crack is present and the local applied stress intensity exceeds the reduced fracture toughness.

3.2.7 Void Swelling and Irradiation Growth

Void swelling is defined as a gradual increase in the volume of a component caused by formation of microscopic cavities in the material. These cavities result from the nucleation and growth of clusters of irradiation produced vacancies. Helium produced by nuclear transmutations can have a significant impact on the nucleation and growth of cavities in the material. Void swelling may produce dimensional changes that exceed the tolerances on a component. Strain gradients produced by differential swelling in the system may produce significant stresses. Severe swelling (>5% by volume) has been correlated with extremely low fracture toughness values. Also included in this description is irradiation growth of anisotropic materials, which is known to cause significant dimensional changes in in-core instrumentation tubes fabricated from zirconium alloys. While the initial aging effect is dimensional change and distortion, severe void swelling may result in cracking under stress.

3.2.8 Thermal and Irradiation-Enhanced Stress Relaxation or Irradiation-Enhanced Creep

The loss of preload aging effect can be caused by the aging mechanisms of stress relaxation or creep. Thermal stress relaxation (or, primary creep) is defined as the unloading of preloaded components due to long-term exposure to elevated temperatures, such as seen in PWR internals. Stress relaxation occurs under conditions of constant strain where part of the elastic strain is replaced with plastic strain. Available data show that thermal stress relaxation appears to reach saturation in a short time (< 1000 hours) at PWR internals temperatures.

Creep (or more precisely, secondary creep) is a slow, time and temperature dependent, plastic deformation of materials that can occur when subjected to stress levels below the yield strength (elastic limit). Creep occurs at elevated temperatures where continuous deformation takes place under constant strain. Secondary creep in austenitic stainless steels is associated with temperatures higher than those relevant to PWR internals even after taking into account gamma heating. However, irradiation-enhanced creep (or more simply, irradiation creep) or irradiation-enhanced stress relaxation (ISR) is an athermal process that depends on the neutron fluence and stress; and, it can also be affected by void swelling should it occur. The aging effect is a loss of mechanical closure integrity (or, preload) that can lead to unanticipated loading which, in turn, may eventually cause subsequent degradation by fatigue or wear and result in cracking.

3.3 Component Categorization and Aging Management Strategy Development Results Summary

3.3.1 Method and Definitions

This sub-section provides a summary of the results of the categorization of PWR internals after the initial screening. In this exercise, Failure Modes, Effects, and Criticality Analyses (FMECA) were applied to the PWR internals. Based upon the FMECA results, the most affected PWR internals were placed into Category C, while the components that are only moderately affected were placed into Category B. In addition, the FMECA process determined that some components not initially Category A were sufficiently unaffected by consequences to be subsequently placed into Category A.

In addition to this categorization using FMECA, a more refined assessment involved functionality assessment of some of the components other than Category A components with the intent to determine the tolerance of components and systems of components to aging degradation effects. When the functionality assessments were completed, all PWR internals were placed into four functional groups, as summarized below:

- **Primary:** those PWR internals that are highly susceptible to the effects of at least one of the eight aging mechanisms were placed in the Primary group. The aging management requirements that are needed to ensure functionality of Primary components are described in these I&E guidelines. The Primary group also includes components which have shown a degree of tolerance to a specific aging degradation effect, but for which no highly susceptible component exists or for which no highly susceptible component is accessible.
- **Expansion:** those PWR internals that are highly or moderately susceptible to the effects of at least one of the eight aging mechanisms, but for which functionality assessment has shown a degree of tolerance to those effects, were placed in the Expansion group. The schedule for implementation of aging management requirements for Expansion components will depend on the findings from the examinations of the Primary components at individual plants.
- **Existing Programs:** those PWR internals that are susceptible to the effects of at least one of the eight aging mechanisms and for which generic and plant-specific existing AMP elements are capable of managing those effects, were placed in the Existing Programs group.
- **No Additional Measures:** those PWR internals for which the effects of all eight aging mechanisms are below the screening criteria were placed in the No Additional Measures group. Additional components were placed in the No Additional Measures group as a result of FMECA and the functionality assessment. No further action is required by these guidelines for managing the aging of the No Additional Measures components.

The categorization and analysis processes described herein are not intended to supersede any ASME B&PV Code Section XI [2] requirements. Any components that are classified as core support structures as defined in ASME B&PV Code Section XI IWB 2500 IWA 9000, and listed in Table IWB 2500-1. Category B-N-3 [2] have requirements that remain in effect and may only be altered as allowed by 10CFR50.55a [4].

3.3.2 Results of Categorization and Aging Management Strategy Development

The results of this process are described below and shown in Tables 3-1 through 3-3. In these tables, the right-hand column characterizes the final group: "P" corresponds to Primary components, "E" corresponds to Expansion components, "X" to Existing Programs components and "N" refers to No Additional Measures components. "A", "B" and "C" refers to the categories after the initial screening and FMECA.

- Of the total components identified for the B&W-designed PWR internals [8], the 39 components listed in Table 3-1 were determined to require further evaluation. Of these, 14 are Primary components and 12 are Expansion components, with the remaining 13 requiring No Additional Measures. There are no Existing Programs components for the B&W-designed PWR internals.

- Of the total components identified for the CE-designed PWR internals [10], the 28 components listed in Table 3-2 were determined to require further evaluation. Of these, 14 are Primary components, 8 are Expansion components, 3 are Existing Programs components, with the remaining 3 requiring No Additional Measures.
- Of the total components identified for the Westinghouse-designed PWR internals [10], the 32 components listed in Table 3-3 were determined to require further evaluation. Of these, 9 are Primary components, 9 are Expansion components, 8 are Existing Programs components, with the remaining 6 requiring No Additional Measures.

Table 3-1
Final disposition of B&W internals

Component	Material	Initial Category	SCC	IASCC	Wear	Fatigue	TE	IE	VS	ISR and IC	Final Group
Plenum Cover Assembly											
Plenum Cover Weldment Rib Pads	304 SS	C	A	A	P	A	A	A	A	A	P
Plenum Cover Support Flange	304 SS	C	A	A	P	A	A	A	A	A	P
Alloy X-750 Dowels-to-Plenum Cover Bottom Flange Welds	Alloy 82 Weld	B	N	A	A	A	A	A	A	A	N
Control Rod Guide Tube (CRGT) Assembly											
CRGT Spacer Castings	CF3M	B	A	A	A	A	P Note 1	A	A	A	P Note 1
CRGT Rod Guide Tubes	304L SS	B	A	A	N	A	A	A	A	A	N
CRGT Rod Guide Sectors	304L SS	B	A	A	N	A	A	A	A	A	N
Core Support Shield Assembly											
CSS Top Flange	304 SS	C	A	A	P	A	A	A	A	A	P
UCB Bolts	Alloy A-286 or Alloy X-750	C	P	A	A	A	A	A	A	A	P
CSS Cast Outlet Nozzles (ONS-3, DB)	CF8	B	A	A	A	A	A Note 2	A	A	A	N Note 2
CSS Vent Valve Top Retaining Ring	15-5PH	B	A	A	A	A	P	A	A	A	P
CSS Vent Valve Bottom Retaining Ring	15-5PH	B	A	A	A	A	P	A	A	A	P

Table 3-1
Final disposition of B&W internals (continued)

Component	Material	Initial Category	SCC	IASCC	Wear	Fatigue	TE	IE	VS	ISR and IC	Final Group
Core Barrel Assembly											
Core Barrel Cylinder (Including Vertical and Circumferential Seam Welds)	304 SS, 308L SS Welds	B	A	A	A	A	A	E	A	A	E
Alloy X-750 Core Barrel-to-Former Plate Dowel	Alloy X-750	B	N	A	A	A	A	N	A	A	N
Alloy X-750 Dowel-to-Core Barrel Cylinder Fillet Welds	Alloy 82 Weld	B	N	A	A	A	A	A	A	A	N
Thermal Shield Upper Restraint Cap Screws (Not Exposed)	304 SS	B	A	A	N	N	A	A	A	N	N
Baffle Plates	304 SS	C	A	N	A	A	A	P	N	A	P
Former Plates	304 SS	C	A	N	A	A	A	E	N	A	E
CB Bolts	304 SS	C	A	E	E	E	A	E	N	E	E
FB Bolts (Note 3)	304 SS	C	A	P	P	P	A	P	N	P	P
Internal BB Bolts (Note 3)	304 SS	C	A	N	E	E	A	E	N	E	E
External BB Bolts	304 SS	C	A	E	E	E	A	E	N	E	E
Accessible Locking Device and Locking Weld (FB Bolts and Internal BB Bolts)	304 SS Locking Device, 308L SS Locking Weld	B	A	P	A	A	A	P	A	A	P
Inaccessible Locking Device and Locking Weld (CB Bolts and External BB Bolts)	304 SS Locking Device, 308L SS Locking Weld	B	A	E	A	A	A	E	A	A	E

Table 3-1
Final disposition of B&W internals (continued)

Component	Material	Initial Category	SCC	IASCC	Wear	Fatigue	TE	IE	VS	ISR and IC	Final Group
LCB Bolts	Alloy A-286 or Alloy X-750	C	P	A	A	A	A	A	A	A	P
UTS Bolts	Alloy A-286 or Alloy X-750	B	E	A	A	A	A	A	A	A	E
SSHT Studs/Nuts (CR-3) or Bolts (DB)	Alloy X-750	B	E	A	A	A	A	A	A	A	E
Upper Grid Assembly											
Alloy X-750 Dowel-to-Upper Grid Rib Section Bottom Flange Welds	Alloy 82 Weld	B	N	A	A	A	A	A	A	A	N
Upper Fuel Assembly Support Pads: Alloy X-750 Dowel Locking Weld	Alloy 82 Weld	B	E	A	A	A	A	A	A	A	E
Lower Grid Assembly											
Lower Fuel Assembly Support Pads: Pad, Pad-to-Rib Section Weld, Alloy X-750 Dowel, Cap Screw, Their Locking Welds	304SS with 308L SS Weld, Except Alloy X-750 Dowel with Alloy 69 Weld	B	A or E Note 4	A	A	A	A	E	A	A	E
Lower Grid Assembly Alloy X-750 Dowel-to-Guide Block Welds	Alloy 82 Weld	B	P	A	A	A	A	A	A	A	P
Alloy X-750 Bolts for Lower Grid Shock Pads (TMI-1 only)	Alloy X-750	B	E	A	A	A	A	A	A	A	E
Alloy X-750 Dowel-to-Lower Grid Shell Forging Welds	Alloy 82 Weld	B	N	A	A	A	A	A	A	A	N
Alloy X-750 Dowel-to-Lower Grid Rib Section Welds	Alloy 69 Weld	B	N	N	A	A	A	N	A	A	N

Table 3-1
Final disposition of B&W internals (continued)

Component	Material	Initial Category	SCC	IASCC	Wear	Fatigue	TE	IE	VS	ISR and IC	Final Group
Lower Grid Rib-to-Shell Forging Cap Screws	304 SS	B	A	A	N	N	A	A	A	N	N
Lower Grid Support Post Pipe Cap Screws	304 SS	B	A	A	N	N	A	A	A	N	N
LTS Studs/Nuts or Bolts	Alloy A-286 or Alloy X-750	B	E	A	A	A	A	A	A	A	E
Flow Distributor Assembly											
FD Bolts	Alloy A-286 or Alloy X-750	C	P Note 5	A	A	A	A	A	A	A	P Note 5
Alloy X-750 Dowel-to-Flow Distributor Flange Welds	Alloy 82 Weld	B	N	A	A	A	A	A	A	A	N
IMI Guide Tube Assembly											
IMI Guide Tube Spiders and Spider-to-Lower Grid Rib Section Welds	CF8, 308L SS Weld	B	A	A	A	A	P	P	A	A	P

Notes to Table 3-1:

1. The CRGT spacer castings upgraded to "Primary" from "Expansion" due to deletion of previously linked "Primary" components (CSS vent valve discs and CSS vent valve shaft/hinge pins) which are active components and thus not subject to aging management requirements.
2. Thermal Embrittlement (TE) revised from "P" to "A" based on review of material data (ferrite content) from ONS-3 and DB; final grouping accordingly changed from "Primary" to "No Additional Measures"
3. Bolt overload after hard contact with the baffle and former plates is identified in Reference 13. This mechanism is only applicable to the FB bolts and internal BB bolts and their locking devices; "Primary" for the FB bolts and their locking devices, and "Expansion" for the internal BB bolts and their locking devices.
4. Only the Alloy X-750 dowel locking weld in the listed items for the lower fuel assembly support pads is susceptible to SCC and categorized as Expansion for SCC. Other items are Category A for SCC.
5. Upgraded to "Primary" from "Expansion" in accordance with the NRC SER [27].

Table 3-2
Final disposition of CE internals

Component	Material	Initial Category	SCC	IASCC	Wear	Fatigue	TE (Note 1)	IE (Note 1)	VS	ISR and IC	Final Group
Upper Internals Assembly											
Fuel Alignment Plate (Core Shrouds with Full-Height Shroud Plates)	304 SS	B	N	A	N	P	A	A	A	A	P
Lower Support Structure											
Core Support Plate	304 SS 304L SS	C	N	N	N	P	A	P	A	A	P
Fuel Alignment Pins (Core Shrouds with Full-Height Shroud Plates)	A286 SS	C	A	X	X	X	A	X	A	X	X
Core Support Columns	304 SS	B	P Note 4	P Note 4	A	P Note 4	A	P Note 4	A	A	P Note 4
Core Support Columns	CF8	B	P Note 4	P Note 4	A	P Note 4	P Note 4	P Note 4	A	A	P Note 4
Core Support Deep Beams (Core Shrouds with Full-Height Shroud Plates)	304 SS	C	X	X	A	P	A	P	A	A	P
Core Support Column Bolts	316 SS	B	A	E	N	E	A	E	A	N	E
Lower Core Support Beams	304 SS	A	E Note 5	A	A	E Note 5	A	A	A	A	E Note 5
Control Element Assembly (CEA)-Shroud Assemblies											
Instrument Tubes	304 SS	B	P	A	A	P	A	A	A	A	P
Core Support Barrel Assembly											
Upper Cylinder (including girth welds)	304 SS	B	E	A	A	A	A	A	A	A	E

Table 3-2
Final disposition of CE internals (continued)

Component	Material	Initial Category	SCC	IASCC	Wear	Fatigue	TE (Note 1)	IE (Note 1)	VS	ISR and IC	Final Group
Lower Cylinder Girth Welds	304 SS	C	P Note 4	P Note 6	A	A	A	P Note 4	A	A	P Note 4
Lower Cylinder Axial Welds	304 SS	C	E	E	A	A	A	E	A	A	E
Upper Core Barrel Flange Weld	304 SS	B	P	A	X	A	A	A	A	A	P
Upper Core Barrel Flange	304 SS	A	E Note 5	N	A	N	N	N	N	N	E Note 5
Lower Core Barrel Flange	304 SS	B	E	A	A	E	A	A	A	A	E
Lower Core Barrel Flange Weld	304 SS	B	E	A	A	P	A	A	A	A	P
Thermal Shield Positioning Pins (Note 2)	UNS S21800	B	A	A	N	N	A	A	A	N	N
Core Shroud Assembly											
Shroud Plates (Bolted) (Entire Assembly)	304 SS	C	N	E	A	A	A	P	P	A	P
Shroud Plates (Welded)	304 SS	C	N	P	A	A	A	P	P	A	P
Former Plates (Bolted) (Entire Assembly)	304 SS	B	N	E	A	A	A	P	P	A	P
Former Plates (Welded)	304 SS	B	N	P	A	A	A	P	P	A	P
Ribs	304 SS	B	N	E	A	A	A	E	N	A	E
Rings (Core Shrouds with Full-Height Shroud Plates)	304 SS	B	N	E	A	A	A	E	N	A	E
Core Shroud Bolts	316 SS	B	A	P	N	N	A	P	P	P	P
Barrel-Core Shroud Bolts	316 SS	B	A	E	N	N	A	E	A	E	E
Core Shroud Tie Rods	348 SS	B	A	A	N	N	A	N	A	N	N
Core Shroud Tie Rod Nuts	316 SS	B	A	A	N	N	A	N	A	N	N

Table 3-2
Final disposition of CE internals (continued)

Component	Material	Initial Category	SCC	IASCC	Wear	Fatigue	TE (Note 1)	IE (Note 1)	VS	ISR and IC	Final Group
Guide Lug Insert Bolts (Note 3)	A286 SS	B	A	A	X	X	A	A	A	X	X
In-Core Instrumentation (ICI)											
ICI Thimble Tubes-Lower	Zircaloy-4	C	A	A	X	A	A	A	A	A	X

Notes to Table 3-2:

1. The significance of thermal and irradiation embrittlement is directly related to the probability of a flaw existing in the component. There are no recommendations for inspection to determine embrittlement level because these mechanisms cannot be directly observed. However, potential embrittlement must be considered in flaw tolerance evaluations.
2. One plant has an existing program for this item.
3. Bolt deterioration may lead to degradation in lug fixtures. Inspection recommendations relate to the entire guide lug fixture.
4. Upgraded to "Primary" from "Expansion" in accordance with the NRC SER [27].
5. Upgraded to "Expansion" from "No Additional Measures" in accordance with the NRC SER [27].
6. Mechanism "IASCC" upgraded from "No Additional Measures" to "Primary" for lower cylinder welds in accordance with the NRC SER [27].

Table 3-3
Final disposition of Westinghouse internals

Component	Material	Initial Category	SCC	IASCC	Wear	Fatigue	TE (Note 1)	IE (Note 1)	VS	ISR and IC	Final Group
Control Rod Guide Tube Assembly											
Lower Flanges	CF8	B	P	A	A	P	P	P	A	A	P
Guide Plates (Cards)	304 SS	C	N	A	P	N	A	A	A	A	P
C-Tubes (Note 2)	304 SS	C	A	A	P	A	A	A	A	A	N
Sheaths (Note 2)	304 SS	C	A	A	P	A	A	A	A	A	N
Guide Tube Support Pins	Alloy X-750	C	X	A	X	X	A	A	A	N	X
Upper Internals Assembly											
Upper Support Ring or Skirt	304 SS	B	E	A	A	X	A	A	A	A	X
Upper Core Plate	304 SS	A	A	A	E Note 5	E Note 5	A	A	A	A	E Note 5
Baffle-Former Assembly											
Baffle-Edge Bolts	316 SS, 347 SS	C	A	P	N	P	A	P	P	P	P
Baffle Plates and Former Plates (Note 3)	304 SS	B	A	N	A	A	A	N	P	A	P
Baffle-Former Bolts	316 SS, 347 SS	C	A	P	N	P	A	P	P	P	P
Barrel-Former Bolts	316 SS, 347 SS	C	A	E	N	E	A	E	E	E	E
Bottom Mounted Instrumentation System											
BMI Column Bodies	304 SS	B	N	N	A	E	A	E	N	A	E
BMI Column Collars	304 SS	B	A	N	A	A	A	N	N	A	N
BMI Column Cruciforms	CF8	B	A	N	A	A	N	N	N	A	N
BMI Column Extension Tubes	304 SS	B	N	N	A	A	A	N	N	A	N
Flux Thimble Tube Plugs	304 SS	B	N	N	A	A	A	N	N	A	N
Flux Thimbles (Tubes)	316 SS	C	N	N	X	A	A	N	N	A	X

Table 3-3
Final disposition of Westinghouse internals (continued)

Component	Material	Initial Category	SCC	IASCC	Wear	Fatigue	TE (Note 1)	IE (Note 1)	VS	ISR and IC	Final Group
Core Barrel Assembly											
Core Barrel Flange	304 SS	B	E	A	X	A	A	A	A	A	X
Core Barrel Outlet Nozzle Welds	304 SS	B	E	A	A	E	A	A	A	A	E
Core Barrel Girth Welds	304 SS	C	P Note 6	P Note 6	A	A	A	P Note 6	A	A	P Note 6
Core Barrel Axial Welds	304 SS	C	E	E	A	A	A	E	A	A	E
Upper Core Barrel Flange Weld	304 SS	C	P	E	A	A	A	A	A	A	P
Lower Internals Assembly											
Lower Core Plate	304 SS	C	N	X	X	X	A	X	N	A	X
XL Lower Core Plate	304 SS	C	N	X	X	X	A	X	A	A	X
Lower Support Casting	CF8	A	A	A	A	A	E Note 5	A	A	A	E Note 5
Lower Support Forging	304 SS	A	A	A	A	A	A	A	A	A	E Note 5
Lower Support Assembly											
Lower Support Column Bodies	CF8	B	A	E	A	A	N	E	N	A	E
Lower Support Column Bodies	304 SS	B	A	E	A	A	A	E	N	A	E
Lower Support Column Bolts	304 SS	B	A	E	N	E	A	E	N	E	E
Thermal Shield Assembly											
Thermal Shield Flexures	304 SS	B	A	N	P	P	A	N	A	N	P
Alignment and Interfacing Components											
Clevis Insert Bolts	Alloy X-750	B	A	A	X	A	A	A	A	A	X
Internals Hold Down Spring (Note 4)	304 SS	B	A	A	P	A	A	A	A	A	P
Upper Core Plate Alignment Pins	304 SS	B	X	A	X	A	A	A	A	A	X

Notes to Table 3-3:

1. The significance of thermal and irradiation embrittlement is directly related to the probability of a flaw existing in the component. There are no recommendations for inspection to determine embrittlement level because these mechanisms cannot be directly observed. However, potential embrittlement must be considered in flaw tolerance evaluations.
2. Some of the items in the control rod guide tube (CRGT) assembly, namely the C-tubes and sheaths, have been placed in the No Additional Measures group, because decisions on remediation of wear and degradation in the CRGT assembly will be based only on the conditions detected in the Primary CRGT item, the guide tubes (cards).
3. The concern is a result of the collective interaction of all components that comprise the assembly and not strictly focused on the plates.
4. The hold-down spring does not directly degrade by wear. It first degrades by loss in preload, which leads to wear when an inadequate preload remains.
5. Upgraded to "Expansion" from "No Additional Measures" in accordance with the NRC SER [27].
6. Upgraded to "Primary" from "Expansion" in accordance with the NRC SER [27].

4 AGING MANAGEMENT REQUIREMENTS

The ultimate goal of an aging management program (AMP) is to monitor the condition of the internals to maintain appropriate levels of plant safety and reliability. Properly managed, the plants will fulfill their license renewal commitments.

Inspection and evaluation in support of aging management requirements typically consists of the following:

- selection of items for aging management;
- selection of the type of examination or other methodologies appropriate for each applicable degradation mechanism;
- specification of the required level of examination qualification;
- schedule of first and frequency of any subsequent examinations;
- sampling and coverage;
- expansion of scope if sufficient evidence of degradation is observed;
- examination acceptance criteria;
- methods for evaluating examination results not meeting the examination acceptance criteria;
- updating the program based on industry-wide results; and
- contingency measure to repair, replace, or mitigate.

The listed elements of inspection and evaluation interrelate. For example, the particulars of the examination acceptance criteria may affect the rules for sampling or frequency of examination.

This section of the guidelines specifies aging management requirements that are appropriate to detect the expected effects of the degradation mechanisms, and are considered acceptable for the development of an AMP. The criterion for acceptability of an aging management requirement is that it accomplishes the AMP goal, namely, ensuring the continued achievement of safety related and economically important functions of the internals. The technical bases used to develop these aging management requirements are documented [13, 14].

Some of the aging management requirements listed, for example, examination acceptance criteria, deserve greater elaboration and are therefore discussed in Section 5.

Section 4.1 describes the overall aging management approach. Then, Section 4.2 describes the various examination methodologies, ranging from general condition visual examinations to more rigorous visual, surface, and volumetric examinations, with a final sub-section that describes physical measurement. Section 4.3 summarizes the examination requirements that are recommended for two groups of PWR internals – Primary and Expansion.

The requirements stated within this section may revert to those required by ASME Code Section XI [2] if components are repaired, modified or replaced such that the effects of aging are fully mitigated. Demonstration of the adequacy of repair, replacement, or modification activities to fully mitigate the effect of aging is the responsibility of the owner. In addition, repair, replacement or modification activities may also warrant revision to the scope and/or frequency of the generic requirements stated in these guidelines. This includes re-establishing the technical basis for the replaced components (if not fully mitigated) and the technical basis for examination of any linked Expansion components, which was developed on the basis of expert panel solicitation [15]. Individual utilities will be responsible for the technical justification of such activities to demonstrate their acceptability for different requirements than those stated in these guidelines.

The requirements for the PWR internals in the Existing Programs group are described in Section 4.4. As described in Section 4.5, those PWR internals in the No Additional Measures group require no further actions with respect to management of aging degradation, other than to continue any existing requirements that affect these components.

4.1 Aging Management Approach

The aging management approach for PWR internals consists of four major elements: (1) component categorization and aging management strategy development; (2) selection of aging management methodologies for PWR internals that are both appropriate and based on an adequate level of applicable experience; (3) qualification of the recommended methodologies that is based on adequate technical justification; and (4) implementation of the recommendations based on the Guideline for the Management of Materials Issues [1]. Each element in the approach is described in greater detail in the following paragraphs.

4.1.1 PWR Internals Categorization and Aging Management Strategy Development

The PWR internals categorization and aging management strategy development were summarized in Section 3.

4.1.2 Selection of Established Aging Management Methodologies

The second part of the aging management approach involved the selection of aging management methodologies for the PWR internals. The criteria for selection were based on:

- the methodologies should be appropriate for the characterization of particular age-related degradation effects; and
- the aging management methodologies should concentrate on techniques that have been subject to widespread application.

For these two reasons, the selected aging management methodologies emphasize existing, well-proven techniques that have been subject to widespread, relevant application. These methodologies are described in Section 4.2.

4.1.3 Aging Management Methodology Qualification

An extensive experience base for the aging management methodologies described in this section of the I&E guidelines permits selection of known aging management methodologies. Many inspections specified herein are remote visual examinations, whether visual VT-1, EVT-1 or VT-3. For remote visual examinations, no procedural qualifications are required beyond ASME B&PV Code Section XI requirements. Remote visual examinations must meet the additional generic requirements of the Inspection Standard [3] for equipment and training of personnel, and in the case of visual EVT-1, a surface condition assessment and limitations on camera angle and scan speed. All other methodologies specified herein already have well established procedural qualifications, such as volumetric examination of bolting [16]. Thus the level of procedural qualification for examinations other than remote visual is limited to technical justification. This level of qualification is appropriate. Failures of internals do not result in pressure boundary failures. Internals are either of robust design resulting in flaw tolerance well above the detection level that can be established via technical justification or consist of assemblies for which a single (or often multiple) component item failure does not prevent the assembly from performing its function.

The Inspection Standard [3] provides detailed guidance for conducting and justifying the selected examination techniques and the technical justifications required for different examination methodologies and component configurations.

4.1.4 Implementation of Aging Management Requirements

Information on the implementation of the aging management requirements is provided in Section 7 of these I&E guidelines.

4.2 Aging Management Methodologies

The aging management methodologies described in these guidelines include visual examinations, surface examinations, volumetric examinations, and physical measurements. Each of these methodologies is suitable for managing the effects of one or more aging degradation mechanisms for PWR internals, depending upon:

- tolerance of the component functionality to the progression of particular effects;
- accessibility of the component by the equipment needed for the examination; and
- suitability of the equipment for detecting the particular effect.

Where appropriate the examination methodologies selected for use in these guidelines are as specified in the latest U.S. Nuclear Regulatory Commission (NRC) approved edition and addenda of ASME B&PV Code Section XI [2], including those discussed in 4.2.1 and 4.2.2.

These methodologies are described in the following sub-sections.

4.2.1 Visual (VT-3) Examination

One examination methodology selected for use in these guidelines, which has an extensive history of use for PWR internals, is visual (VT-3) examination. Such visual examinations are exclusively relied upon for detection of general degradation of PWR internals subject to Table IWB-2500-1 B-N-3 [2] requirements. Visual (VT-3) examinations are conducted to determine the general mechanical and structural condition of components by detecting discontinuities and imperfections, such as loss of integrity at bolted or welded connections, loose or missing parts, debris, corrosion, wear, or erosion; and by identifying conditions that could affect operational or functional adequacy of components. This type of examination has been determined to be acceptable for the continued monitoring of many of the internals within the scope of these guidelines.

When specified in these guidelines, a visual (VT-3) examination is conducted in accordance with the requirements of the Inspection Standard [3]. Visual (VT-3) examinations of internals are conducted using remote examination techniques, because of personnel radiation exposure issues.

A large amount of industry experience is available relative to the application of visual (VT-3) examination procedures for examining PWR internals; however, implementation of character height requirements for VT-3 is relatively new. Thus the VT-3 required by these guidelines has greater detection capability than most of the Table IWB-2500-1 B-N-3 [2] examinations previously conducted.

4.2.2 Visual (VT-1 and EVT-1) Examinations

Other examination methodologies selected for use in these guidelines are visual (VT-1 and EVT-1) examinations. The visual (VT-1) examination and the enhanced visual (EVT-1) examination were selected where a greater degree of detection capability than visual (VT-3) examination was needed to manage the aging effect. Unlike the detection of general degradation conditions by visual (VT-3) examination, visual (VT-1) and enhanced visual (EVT-1) examinations are conducted to detect discontinuities and imperfections on the surface of components, including such conditions as cracks, wear, corrosion, or erosion. Specifically, VT-1 is used for the detection of surface discontinuities such as gaps, while EVT-1 is used for the detection of surface breaking flaws.

When specified in these guidelines, a visual (VT-1) examination is conducted in accordance with the requirements of the Inspection Standard [3]. Enhanced visual (EVT-1) examination is also conducted in accordance with the requirements described for visual (VT-1) examination with additional requirements (such as camera scanning speed) as specified in the Inspection Standard [3].

As with visual (VT-3) examination, the current ASME B&PV Code [2] requirements for visual (VT-1) examination became more rigorous than the previous ASME B&PV Code versions. Many previous VT-1 examinations were only required to discern a 1/32" black line on a gray background. These limitations led the NRC and industry to adopt modified visual examinations for use in detecting flaws discovered in boiling water reactor (BWR) internals. The most recent research conducted by the EPRI Non-Destructive Examination (NDE) Center established the VT-1 character heights specified in Reference 2 as equally or better able to detect the degradation effects than the modified visual examination requirements developed previously [17].

4.2.3 Surface Examination

In order to further characterize discontinuities on the surface of components, surface examination can supplement either visual (VT-3) or (VT-1/EVT-1) examinations specified in these guidelines. This supplemental examination may thus be used to reject or accept relevant indications. A surface examination is an examination that indicates the presence of surface discontinuities, and the ASME B&PV Code [2] lists magnetic particle, liquid penetrant, eddy current, and ultrasonic examination methods as surface examination alternatives. Here, only the electromagnetic testing (ET), also called eddy current surface examination method, is covered.

When selected for use as a supplemental examination to examinations performed in these guidelines, an ET examination is conducted in accordance with the requirements of the Inspection Standard [3].

ET examination is widely used for heat exchanger tubing inspections. Eddy currents are induced in the inspected object by electromagnetic coils, with disruptions in the eddy current flow caused by surface or near-surface anomalies detected by suitable instrumentation. Industry experience with ET examination is relatively robust, especially in the aerospace and petroleum refinery industries. The experience base for PWR nuclear systems is moderately robust, in particular for examination of steam generator, flux thimble, and heat exchanger tubing.

4.2.4 Volumetric Examination

Another methodology selected for use in these guidelines is volumetric examination. An ultrasonic examination (UT) was selected where visual or surface examination is unable to detect the effect of the age-related degradation for some PWR internals. For example, irradiation-assisted stress corrosion cracking (IASCC) in baffle/former bolts may occur under the bolt head – in the shank or threaded region – and will be undetectable by visual or surface examination unless the bolt is removed and subject to examination over its entire length.

When specified in these guidelines, an ultrasonic examination (UT) is conducted in accordance with the requirements of the Inspection Standard [3].

While UT has only been selected for use in these guidelines for detection of aging effects in bolting, UT is also permissible as an alternative or supplement to the specified visual examinations for other configurations such as plates and welds. This is consistent with Reference 2.

The industry has had extensive experience with the application of ultrasonic examination (UT) to PWR internals bolts, pins, and fasteners, in particular with baffle/former bolting examinations. The industry also has extensive experience in applying UT to BWR internals to detect intergranular stress corrosion cracking (IGSCC) in stainless steel and nickel-base welded plates, stainless steel internals piping, and nickel-base forgings and bolting.

4.2.5 Physical Measurements

The effects of loss of material caused by wear, the loss of pre-load or clamping force caused by such mechanisms as thermal and irradiation-enhanced stress relaxation, and excessive distortion or deflection caused by void swelling can be managed in some cases by physical measurements. Satisfaction of prescribed limits on these physical measurements (see Section 5.2) is intended to demonstrate that the affected components remain functional and can continue in service for a determined period until the next set of physical measurements. If the prescribed limits are exceeded, corrective action or evaluation for continued service is required.

In some cases, these effects may involve changes in clearances, settings, and physical displacements that can be monitored by visual means, supplemented by physical measurements that characterize the magnitude of the effects. This methodology may be used in conjunction with visual (VT-3) examination, which includes "verifying parameters, such as clearances, settings, and physical displacements." The measurement of these parameters and their comparison to prescribed limits extends beyond visual (VT-3) examination, and will be referred to as "physical measurement of the effects of degradation."

4.3 Primary and Expansion Component Requirements

The aging management requirements for Primary and Expansion PWR internals are covered in this section. As described in Section 3.3, Primary components are those for which the effects of at least one of the eight aging mechanisms is above the screening criteria, and for which additional aging management is needed to manage those effects. The particular additional aging management methodologies were selected from the methodologies described in Section 4.2. The implementation schedule for the Expansion components will depend on the findings from the application of the additional aging management methodologies to the Primary components. The expansion criteria are defined in Section 5.

Sections 4.3.1, 4.3.2, and 4.3.3 identify and discuss the aging management methodologies for the Primary and Expansion components for B&W, CE, and Westinghouse plants, respectively. The requirements for these components are listed in Tables 4-1 through 4-6. For example, the Primary and Expansion requirements for Westinghouse internals are listed in Tables 4-3 and 4-6. These tables contain columns describing the component; any particular applicability requirement for that component; the degradation effect to be detected; the examination method; the examination coverage; and any linkage between the Primary and Expansion components. The technical bases for the examination requirements are contained in the aging management strategy reports [13, 14].

There are no specified examinations where inadequate coverage is anticipated to be an issue. However if a utility determines that the examination coverage is questionable with respect to meeting the intent of the guidelines, the condition should be entered in the utility's corrective action program for disposition.

The term "accessible" as used in Tables 4-1 through 4-6 is defined as a component surface or volume for which an examination is specified in accordance with MRP-228 that can be examined with the technologies specified in MRP-228. This accessibility is consistent with current ASME Section XI practices.

4.3.1 B&W Components

Tables 4-1 and 4-4 describe the examination requirements for PWR internals Primary and Expansion components for B&W plants.

The following is a list of the B&W Primary and Expansion components by examination technique.

- ***Visual (VT-3) Examination***

  Primary (applicable to all plants):

  – Baffle plates

    Expand to:

    - Core barrel cylinder (including vertical and circumferential seam welds)
    - Former plates

Since the regions around flow or bolt holes are preferential crack initiation sites, the surface area within one inch of the flow and bolt hole edges represents the required examination coverage.

Note that even though the core barrel cylinder and the former plates are Expansion components, they require an evaluation and not an inspection.

  Primary (applicable to all plants):

  – Locking devices, including locking welds, of baffle-to-former bolts and internal baffle-to-baffle bolts

    Expand to:

    - Locking devices for the external baffle-to-baffle bolts and barrel-to-former bolts

Note that the bolts associated with the baffle-to-former bolt locking devices are also examined by volumetric (UT) examination. Note also that the locking devices for the internal and external baffle-to-baffle bolts and barrel-to-former bolts require an evaluation and not an inspection.

  Primary (applicable to all plants):

  – Alloy X-750 dowel-to-guide block welds

    Expand to:

    - Alloy X-750 dowel locking welds to the upper and lower grid fuel assembly support pads

The locking welds may be susceptible to cracking as a result of stress corrosion cracking (i.e., primary water stress corrosion cracking (PWSCC)). The recommended program to manage cracking of the locking welds is in conjunction with the existing Examination Category B-N-3 of the ASME B&PV Code Section XI [2] ISI program. The guide block area is accessible when the core support assembly is removed from the vessel. The 10-year interval is considered adequate due to the low consequences of failure. Due to weld residual stresses and the constrained geometry, it is anticipated that significant cracks will be accompanied by locking device/weld separation and therefore be detectable by the visual (VT-3) examination method.

Primary (applicable to all plants):

- IMI guide tube spiders
- IMI guide tube spider-to-lower grid rib section welds

  Expand to:

  - Lower grid fuel assembly support pad items: pad, pad-to-rib section welds, Alloy X-750 dowel, cap screw, and their locking welds

The IMI guide tube spiders and their associated welds, and the lower grid fuel assembly support pads and their associated welds may have degradation by thermal or irradiation embrittlement. The effects of thermal and irradiation embrittlement can be detected by inspection to detect fracture in the items.

The IMI guide tube spiders are primary items for thermal embrittlement.

The lower grid fuel assembly support pad items (Figure 4-6) consist of the stainless steel block, Alloy X-750 dowels, and stainless steel cap screws, all susceptible to irradiation embrittlement. The primary item is the IMI guide tube spider and associated fillet welds. Cracking of the dowel or cap screw tack weld may be observed, but more likely, the aging mechanism will be detected by the grid pad not being properly located. The lower grid fuel assembly support pads and their associated welds are part of the ASME B&PV Code Section XI [2] 10-year ISI program and are inspected via visual (VT-3) examination.

Primary (applicable to all plants):

- CRGT spacer castings

There are no Expansion items for these components.

The control rod guide tube (CRGT) spacer castings (Figure 4-5) are primary items for thermal embrittlement. The effects of thermal embrittlement can be detected by inspection to detect fracture in the CRGT spacer castings. The spacer castings are a part of the CRGT structure. The spacer castings do have limited accessibility from the top or bottom of the CRGT through a center free path. This of course presumes that the plenum assembly is removed from the vessel. Remote video can be used to perform a visual (VT-3) examination at the quarter points where the threaded connections are present. These lanes are not blocked by the rod guide tubes. The examination would look for fracture of the spacer surface or evidence that the spacer is not approximately centered. The threaded fasteners are welded to the OD of the pipe column, so it is possible that a degraded threaded location would not be detected. In this case, it is assumed that the redundant support is acceptable for continued operation.

Primary (applicable to all plants):

- CSS vent valve top retaining ring
- CSS vent valve bottom retaining ring

There are no expansion items for these components.