remaining core barrel welds and the non-cast lower support column bodies in Westinghouse plants. The original degradation effect of concern was cracking from SCC (because of the residual stresses in the welds), but cracking from IASCC – in particular for the circumferential weld connecting the upper and lower core barrel sections that is located adjacent to the core active region – was also a potential concern. This potential concern about IASCC cracking did not extend to the upper core barrel flange weld, which is outside the core active region with relatively low irradiation exposure. However, the results of the functionality analysis showed that irradiation-induced stress relaxation limited the IASCC ratio to 0.41, well below the threshold of concern. Therefore, because of its greater thickness and relatively residual and operating stresses, the upper core barrel flange weld was designated as the Primary component, with the remaining core barrel welds (circumferential and axial) were relatively less affected, becoming designated as Expansion components.

Since these welds are part of a core support structure, they are all subject to ASME Code Section XI Examination Category B-N-3 visual (VT-3) inspections. However, in order to determine the adequacy of those visual examinations for detecting SCC, the experts reviewed available information on the flaw tolerance of structures similar to the core barrel that were known to have reduced fracture toughness from neutron irradiation exposure, including but not limited to the information contained in MRP-210, "Materials Reliability Program: Fracture Toughness Evaluation of Highly Irradiated PWR Stainless Steel Components, June 2007. These and other flaw tolerance calculations have been based on lower bound fracture toughness information and simple geometries – such as a through-wall crack in a flat plate – with remote tensile stress treated parametrically. The experts shared their flaw tolerance information, which showed that critical flaw lengths were two inches or greater for remote tensile stresses of the order of 30 ksi and through-wall flaws. In addition, the experts examined the available information from the functionality analyses on the irradiation-induced stress relaxation at all elevations for the core barrel welds. Some experts argued for looking beyond through-wall flaws to examine critical flaw lengths for part-through flaws. In such cases, surface-breaking flaws greater than five inches in length and extending to over ten inches in length, depending upon flaw depth, were needed to reach critical flaw length. However, the consensus of the experts was to use a conservative expansion criterion of a two-inch-long flaw length for a surface-breaking detected flaw. The experts also debated the need for increasing the rigor of the examination from a VT-3 to a VT-1 and perhaps even to a EVT-1 examination. However, expert judgments on the potential crack-opening surface displacement for a two-inch-long, surface-breaking flaw led to a consensus that the character recognition requirements for the VT-1 examination would be sufficient to ensure detection and length sizing. Finally, the experts debated the need for altering the ASME Code frequency of examination from ten years to some shorter period.

Information from the functionality analyses showed that the conservatisms embedded in the high-leakage and low-leakage core histories, plus the conservatism in the lower-bound flaw tolerance calculations, plus the conservatism in the expansion criterion relative to critical flaw lengths, plus the conservatism of the VT-1 examination were sufficient to warrant continuing the existing ASME Code inspection periodicity.

Another example, the Westinghouse core barrel, is discussed below. The reasoning for linking SCC and IASCC in the primary and expansion strategy for the Westinghouse core barrel is outlined in Section 4.2.2 of MRP-232. The following paragraphs are excerpted from that document.

> *The aging degradation mechanisms identified for the core barrel structure are listed in Table 4-8. Due to the large size of the core barrel and the significance of the welds, sections of the core barrel were originally listed as separate components. Although this division was helpful in the identification of aging degradation issues, for the evaluation of the core barrel, the welded structure will be considered as a single assembly consistent with the approach used for CE-designed plants.*
>
> *By the conventions used in this program, IASCC is defined as the form of SCC that is observed in materials with neutron fluences greater than 3 dpa. Because the core barrel contains both irradiated and unirradiated welds, the core barrel assembly was screened in for both SCC and IASCC. The welds in the core barrel were originally identified as potentially susceptible to SCC due to the residual stresses produced by welding in conjunction with deadweight loads and operational stresses.*

Analysis indicates that irradiation-induced stress relaxation reduces the weld residual stresses below the threshold for IASCC in the section of the core barrel immediately adjacent to the core. However, for core barrel welds outside the active core region, there is no mechanism for stress relaxation. Due to the relatively low potential for reaching or exceeding the IASCC susceptibility ratio, the core barrel welds are not considered to be a lead item for IASCC.

The lack of any known predictive model (or data) for SCC in non-irradiated stainless steels in PWR environments makes it difficult to provide an analysis that eliminates the concern for SCC. The potential for large residual stresses in the unirradiated core barrel welds make them a potential lead component for SCC. Under normal operating conditions the upper flange weld is expected to experience the highest stress. Given the critical structural role of the core barrel, periodic inspection for cracking of the high stress weld is recommended.

Similar types of discussion could be added for each and every Primary to Expansion link. However, the expert elicitation process does not generally lend itself to a detailed narration of the discussion and decision-making process. The notes that were included in "Letter to Reactor Internals Focus Group from MRP, Subject: *Minutes of the Expert Panel Meetings on Expansion Criteria for Reactor Internals I&E Guidelines*, MRP 2008-

036 (via email), June 12, 2008)" give the results and some of the reasoning that took place within the expert panels. In conclusion it can be stated that in no case is a component item in the expansion group predicted to experience the aging effect sooner or at a faster rate than its linked Primary component.

RAI 4-27: MRP-227 and supporting reports do not clearly document how the consideration of degradation mechanisms associated with weld heat-affected zones, weld repair, and variability in welding processes and parameters was addressed in the susceptibility evaluation. Please provide an overview of how these issues were evaluated to determine the final AMP recommendations for welded components and also provide specific examples to illustrate the impact of these issues on the final inspection requirements.

Response: The treatment of welds, including weld heat-affected zones, weld repair, and variability in welding processes and parameters, was not treated identically in the two susceptibility screening efforts, as documented in MRP-189 and MRP-191.

The original version of MRP-189 screened for multi-pass welds, without consideration of welding process or welding geometry, primarily looking for SCC susceptibility. However, Revision 1 of MRP-189 contained an extensive update of the original MRP-189, with the specific intent of addressing multi-pass welds, their heat-affected zones, the variability in welding processes, and any influence of weld geometry. Table 3-2 is similar to the table in the original MRP-189, showing only whether or not the particular internals component contained a multi-pass weld, implying heat-affected zones that were considered to be susceptible to SCC. Revision 1 of MRP-189 contains Section 3.3 and Table 3-3 that address such items as welding process and welding geometry.

Generally, the susceptibility evaluations documented in MRP-191 for Combustion Engineering and Westinghouse internals only divided welds into a separate category for evaluation when the screening criteria of MRP-175 were also separated. For example, the second paragraph in Section 3.2 of MRP-191 cites austenitic stainless steel welds, especially those with less than 5% ferrite content, highly-constrained welds, and parts with > 20% cold work as items for which SCC could be an issue. Then, Table 3-1 of MRP-191 specifically identifies welds separately from other high effective stress locations for SCC screening criteria, while Table 3-2 of MRP-191 implies that welds are included for IASCC through the criterion that all components with effective stresses above 30 ksi were screened in. Table 3-5 in MRP-191 identifies welds separately for thermal aging embrittlement criteria, while Tables 3-6 and 3-7 lump austenitic welds and austenitic base metal in the same category. In other words, welds and their heat-affected zones were treated separately where screening criteria for susceptibility supported such

distinctions. As a result, welds and the volume of material near welds are called out for special attention when susceptibility is so indicated. MRP-191 did not explicitly address variability in welding processes and parameters.

The potential for weld repair (grinding out a defect found in a weld during either pre-service or in-service examination, and re-welding) was treated in both MRP-190 and MRP-191 through the explicit conservative inclusion of welds for SCC or the implicit conservative inclusion of high effective stress locations for IASCC. Such conservative inclusions of weld locations avoided the need for an exhaustive review of component fabrications records, which may or may not have included the level of detail that would have been required to identify specific component locations of concern. The more efficient approach was to assume that all components that were judged to be heavily deformed or welded during manufacture were initially screened in for SCC, regardless of stress level, with a relatively similar conservative approach (see Figure 5-1 in MRP-191) used for initial screening for IASCC. In that way, any potential weld repairs would be captured in the initial screening.

The potential for repair welding of non-welded material, such as to repair porosity in a stainless steel casting was considered outside the scope of the screening exercise.

The MRP-189, Revision 1, process and the MRP-191 process both led to robust and defendable recommendations for specific weld and heat-affected zone inspection requirements.

APPENDIX A – PROPOSED CHANGES to MRP-227-Rev. 0

As discussed in the meeting between the NRC and the MRP/Industry on 10/14/2010, the MRP is proposing some changes to MRP-227-Rev. 0 to the NRC for incorporation into MRP-227-A. The MRP committees have concurred with these changes. All proposed changes are listed below.

1) The MRP proposes to elevate requirement **"7.6 Aging Management Program Results Requirement"** in Section 7 from "**Good practice**" to "**Needed**" and to change the text of this requirement to:

   "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."*

2) The MRP proposes to add a new requirement to Section 7 and to add the following text to Section 7:

   "7.7 Evaluation Requirement

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

3) The MRP proposes to add the following minimum coverage requirements for some of the Primary components.

   - For Table 4-1 (B&W Primary components), notes to the table indicated in quotes will be added to the following components:
     - For Upper core bolts and their locking devices, Lower core bolts and their locking devices, Baffle-to-former bolts, Locking devices including locking welds, or baffle-to-former bolts and internal baffle-to-baffle bolts:

       "A minimum of 75% of the total population (examined + unexamined), including coverage consistent with the Expansion criteria in Table 5-1, must be examined for inspection credit."
   - For Table 4-2 (CE Primary components), notes to the table indicated in quotes will be added to the following components:
     - For Core shroud bolts:

       "A minimum of 75% of the total population (examined + unexamined), including coverage consistent with the Expansion criteria in Table 5-2, must be examined for inspection credit."
     - For Upper (core support barrel) flange weld:

       "A minimum of 75% of the total weld length (examined + unexamined),

including coverage consistent with the Expansion criteria in Table 5-2, must be examined from either the inner or the outer diameter for inspection credit."

- For Table 4-3 (Westinghouse Primary components), notes to the table indicated in quotes will be added to the following components:
    - For Baffle-edge bolts, Baffle-former bolts:

      "A minimum of 75% of the total population (examined + unexamined), including coverage consistent with the Expansion criteria in Table 5-3, must be examined for inspection credit."
    - For Upper core barrel flange weld:

      "A minimum of 75% of the total weld length (examined + unexamined), including coverage consistent with the Expansion criteria in Table 5-3, must be examined from either the inner or the outer diameter for inspection credit."

4) The MRP proposes the following changes to the B&W tables in MRP-227-Rev. 0. Note that these tables do not include all of the proposed changes to the B&W Tables (see point 3 above). A set of tables with all the changes combined could be provided to the NRC later if necessary.

Change a:

As noted in the meeting between the NRC and MRP/Industry on 10/14/2010, AREVA is working with its owners to support implementation of MRP-227. As part of a records review and accessibility evaluation, it was determined that the "CSS vent valve disc shaft or hinge pin" was inaccessible. This component is listed as part of the Core Support Shield Assembly in Table 4-1 of MRP-227, Rev. 0 as shown below.

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 2)	Examination Method/Frequency (Note 2)	Examination Coverage
Core Support Shield Assembly CSS vent valve top retaining ring CSS vent valve bottom retaining ring CSS vent valve disc shaft or hinge pin (Note 1)	All plants	Cracking (TE), including the detection of surface irregularities, such as damaged, fractured material, or missing items	None	Visual (VT-3) examination during the next 10-year ISI. Subsequent examinations on the 10-year ISI interval.	100% of accessible surfaces (See BAW-2248A, page 4.3 and Table 4-1.) See Figures 4-10 and 4-11

In order to reflect the actual generic condition and to clarify the requirements, MRP proposes that the following two rows be inserted into MRP-227-A and the existing row (shown above) be deleted. Also, Figure 4-10 will be revised and be replaced for clarity.

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 2)	Examination Method/Frequency (Note 2)	Examination Coverage
Core Support Shield Assembly CSS vent valve top retaining ring CSS vent valve bottom retaining ring (Note 1)	All plants	Cracking (TE), including the detection of surface irregularities, such as damaged, fractured material, or missing items	None	Visual (VT-3) examination during the next 10-year ISI. Subsequent examinations on the 10-year ISI interval.	100% of accessible surfaces (See BAW-2248A, page 4.3 and Table 4-1.) See Figures 4-10 and 4-11
Core Support Shield Assembly CSS vent valve disc shaft or hinge pin (Note 1)	All plants	Cracking (TE)	None	No examination requirements. Justify by evaluation or by replacement.	Inaccessible. See Figure 4-10.

Change b:

As noted in the meeting between the NRC and MRP/Industry on 10/14/2010, AREVA is working with its owners to support implementation of MRP-227. As part of this effort, it was determined that the "effect (mechanism)" column only identifies the "effect (mechanism)" for the UCB bolts and does not clearly describe the "effect (mechanism)" for the associated locking devices. Also, the associated locking devices were omitted from the "Expansion Link" items. In addition, the locking devices were omitted from the "Examination Coverage" column. This component is listed as part of the Core Support Shield Assembly in Table 4-1 of MRP-227, Rev. 0 as shown below.

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 2)	Examination Method/Frequency (Note 2)	Examination Coverage
Core Support Shield Assembly Upper core barrel (UCB) bolts and their locking devices	All plants	Cracking (SCC)	LCB (Note 3) UTS, LTS, and FD bolts SSHT bolts (CR-3 and DB only) Lower grid shock pad bolts (TMI-1 only)	Volumetric examination (UT) of the bolts within two refueling outages from 1/1/2006 or next 10-year ISI interval, whichever is first. Subsequent examination to be determined after evaluating the baseline results. Visual (VT-3) examination of bolt locking devices on the 10-year ISI interval.	100% of accessible bolts. See Figure 4-7

In order to reflect these above omissions, MRP proposes that the additional wording be inserted into Table 4-1 of MRP-227-A, as shown below.

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 2)	Examination Method/Frequency (Note 2)	Examination Coverage
Core Support Shield Assembly Upper core barrel (UCB) bolts and their locking devices	All plants	Bolt: Cracking (SCC) Locking Devices: Loss of material, damaged, distorted, or missing locking devices (Wear or Fatigue damage by failed bolts)	LCB and their locking devices (Note 3) UTS, LTS, and FD bolts and their locking devices SSHT bolts and their locking devices (CR-3 and DB only) Lower grid shock pad bolts and their locking devices (TMI-1 only)	Volumetric examination (UT) of the bolts within two refueling outages from 1/1/2006 or next 10-year ISI interval, whichever is first. Subsequent examination to be determined after evaluating the baseline results. Visual (VT-3) examination of bolt locking devices on the 10-year ISI interval.	100% of accessible bolts and locking devices. See Figure 4-7.

Change c:

As noted in the meeting between the NRC and MRP/Industry on 10/14/2010, AREVA is working with its owners to support implementation of MRP-227. As part of this effort, it was determined that the "effect (mechanism)" column only identifies the "effect (mechanism)" for the LCB bolts and does not clearly describe the "effect (mechanism)" for the associated locking devices. Also, the associated locking devices were omitted from the "Expansion Link" items. In addition, the locking devices were omitted from the "Examination Coverage" column. This component is listed as part of the Core Support Shield Assembly in Table 4-1 of MRP-227, Rev. 0 as shown below.

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 2)	Examination Method/Frequency (Note 2)	Examination Coverage
Core Barrel Assembly Lower core barrel (LCB) bolts and their locking devices	All plants	Cracking (SCC)	UTS, LTS, and FD bolts SSHT bolts (CR-3 and DB only) Lower grid shock pad bolts (TMI-1 only)	Volumetric examination (UT) of the bolts during the next 10-year ISI interval from 1/1/2006. Subsequent examination to be determined after evaluating the baseline results. Visual (VT-3) examination of bolt locking devices on the 10-year ISI interval.	100% of accessible bolts See Figure 4-8

In order to reflect these above omissions, MRP proposes that the additional wording be inserted into Table 4-1 of MRP-227-A, as shown below.

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 2)	Examination Method/Frequency (Note 2)	Examination Coverage
Core Barrel Assembly Lower core barrel (LCB) bolts and their locking devices	All plants	Bolt: Cracking (SCC) Locking Devices: Loss of material, damaged, distorted, or missing locking devices (Wear or Fatigue damage by failed bolts)	UTS, LTS, and FD bolts and their locking devices SSHT bolts and their locking devices (CR-3 and DB only) Lower grid shock pad bolts and their locking devices (TMI-1 only)	Volumetric examination (UT) of the bolts during the next 10-year ISI interval from 1/1/2006. Subsequent examination to be determined after evaluating the baseline results. Visual (VT-3) examination of bolt locking devices on the 10-year ISI interval.	100% of accessible bolts and locking devices. See Figure 4-8.

Change d:

As noted in the meeting between the NRC and MRP/Industry on 10/14/2010, each of the noted mechanisms in the "Effect (Mechanism)" column for the baffle-to-former bolts in Table 4-1 of MRP-227 Rev. 0 do not result in cracking. AREVA proposes that this column be modified to correctly reflect the effects and age-related degradation mechanisms for this component. This component is listed as part of the Core Barrel Assembly in Table 4-1 of MRP-227, Rev. 0 as shown below.

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 2)	Examination Method/Frequency (Note 2)	Examination Coverage
Core Barrel Assembly Baffle-to-former bolts	All plants	Cracking (IASCC, IE, IC/ISR/Fatigue/Wear, Overload)	Baffle-to-baffle bolts, Core barrel-to-former bolts	Baseline volumetric examination (UT) no later than two refueling outages from the beginning of the license renewal period with subsequent examination after 10 to 15 additional years.	100% of accessible bolts See Figure 4-2

In order to reflect the correct effects and age-related degradation mechanisms for this component, MRP proposes that the following modification be made and note added into Table 4-1 of MRP-227-A to the "Effect (Mechanism)" column shown below.

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 2)	Examination Method/Frequency (Note 2)	Examination Coverage
Core Barrel Assembly Baffle-to-former bolts	All plants	Cracking (IASCC, IE, Overload) (Note 4)	Baffle-to-baffle bolts, Core barrel-to-former bolts	Baseline volumetric examination (UT) no later than two refueling outages from the beginning of the license renewal period with subsequent examination after 10 to 15 additional years.	100% of accessible bolts. See Figure 4-2.

Notes:

1. A verification of the operation of each vent valve shall also be performed through manual actuation of the valve. Verify that the valves are not stuck in the open position and that no abnormal degradation has occurred. Examine the valves for evidence of scratches, pitting, embedded particles, variation in coloration of the seating surfaces, cracking of lock welds and locking cups, jack screws for proper position, and wear. The frequency is defined in each unit's technical specifications or in their pump and valve inservice test programs (see AREVA doc. BAW-2248A, page 4.3 and Table 4-1).

2. Examination acceptance criteria and expansion criteria for the B&W components are in Table 5-1.

3. Expansion to LCB applies if the required Primary examination of LCB has not been performed as scheduled in this table.

4. The primary aging degradation mechanisms for loss of joint tightness for this item are IC and ISR. Fatigue and Wear, which can also lead to cracking, are secondary aging degradation mechanisms after significant stress relaxation and loss of preload has occurred due to IC/ISR. Bolt stress relaxation cannot be readily inspected by NDE. Only bolt cracking is inspected by UT inspection in this table. The effect of loss of joint tightness on the functionality will be addressed by analysis of the core barrel assembly.

Change e:

As noted in the meeting between the NRC and MRP/Industry on 10/14/2010, AREVA is working with its owners to support implementation of MRP-227. As part of this effort, it was determined that the "effect (mechanism)" column only identifies the "effect (mechanism)" for the UTS bolts and SSHT studs/nuts or bolts and does not clearly describe the "effect (mechanism)" for the associated locking devices. Also, the associated locking devices were omitted from the "Item" and "Primary Link" columns. In addition, the locking devices were omitted from the "Examination Method" and "Examination Coverage" columns. This component is listed as part of the Core Barrel Assembly in Table 4-4 of MRP-227, Rev. 0 as shown below.

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method (Note 1)	Examination Coverage
Core Barrel Assembly Upper thermal shield bolts (UTS)	All plants	Cracking (SCC)	UCB and LCB bolts	Volumetric examination (UT)	100% of accessible bolts
Core Barrel Assembly Surveillance specimen holder tube (SSHT) studs/nuts (CR-3) or bolts (DB)	CR-3, DB				See Figure 4-7

In order to reflect these above omissions, MRP proposes that the additional wording be inserted into Table 4-4 of MRP-227-A.

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method (Note 1)	Examination Coverage
Core Barrel Assembly Upper thermal shield bolts (UTS) and their locking devices	All plants	Bolt: Cracking (SCC) Locking Devices: Loss of material, damaged, distorted, or missing locking devices (Wear or Fatigue damage by failed bolts)	UCB and LCB bolts and their locking devices	Bolt: Volumetric examination (UT). Locking Devices: Visual (VT-3) examination.	100% of accessible bolts and locking devices.
Core Barrel Assembly Surveillance specimen holder tube (SSHT) studs/nuts (CR-3) or bolts (DB) and their locking devices	CR-3, DB				See Figure 4-7.

Change f:

As noted in the meeting between the NRC and MRP/Industry on 10/14/2010, AREVA is working with its owners to support implementation of MRP-227. As part of this effort, it was determined that the words—no examination requirements—were omitted from the "Examination Method" column for the core barrel cylinder and former plates items. This component is listed as part of the Core Barrel Assembly in Table 4-4 of MRP-227, Rev. 0 as shown below.

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method (Note 1)	Examination Coverage
Core Barrel Assembly Core barrel cylinder (including vertical and circumferential seam welds) Former plates	All plants	Cracking (IE), including readily detectable cracking	Baffle plates	Justify by evaluation or by replacement.	Inaccessible. See Figure 4-2.

In order to reflect this, MRP proposes that the following wording to be inserted into the "Examination Method" column of Table 4-4 of MRP-227-A.

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method (Note 1)	Examination Coverage
Core Barrel Assembly Core barrel cylinder (including vertical and circumferential seam welds) Former plates	All plants	Cracking (IE), including readily detectable cracking	Baffle plates	No examination requirements. Justify by evaluation or by replacement.	Inaccessible. See Figure 4-2.

Change g:

As noted in the meeting between the NRC and MRP/Industry on 10/14/2010, each of the noted mechanisms in the "Effect (Mechanism)" column for the baffle-to-baffle bolts and core barrel-to-former bolts in Table 4-4 of MRP-227 Rev. 0 do not result in cracking. AREVA proposes that this column be modified to correctly reflect the effects and age-related degradation mechanisms for this component. This component is listed as part of the Core Barrel Assembly in Table 4-4 of MRP-227, Rev. 0 as shown below.

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method (Note 1)	Examination Coverage
Core Barrel Assembly Baffle-to-baffle bolts Core barrel-to-former bolts	All plants	Cracking (IASCC, IE, IC/ISR/Fatigue/Wear, Overload)	Baffle-to-former bolts	Internal baffle-to-baffle bolts: No examination requirements, Justify by evaluation or by replacement.	N/A See Figure 4-2
				External baffle-to-baffle bolts, Barrel-to-former bolts: No examination requirements, Justify by evaluation or by replacement.	Inaccessible See Figure 4-2

In order to reflect the correct effects and age-related degradation mechanisms for this component, MRP proposes that the following modification be made and note added into Table 4-4 of MRP-227-A to the "Effect (Mechanism)" column shown below.

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method (Note 1)	Examination Coverage
Core Barrel Assembly Baffle-to-baffle bolts Core barrel-to-former bolts	All plants	Cracking (IASCC, IE, Overload) (Note 2)	Baffle-to-former bolts	Internal baffle-to-baffle bolts: No examination requirements. Justify by evaluation or by replacement.	N/A. See Figure 4-2.
				External baffle-to-baffle bolts, Barrel-to-former bolts: No examination requirements. Justify by evaluation or by replacement.	Inaccessible. See Figure 4-2.

Note:

1. Examination acceptance criteria and expansion criteria for the B&W components are in Table 5-1.

The primary aging degradation mechanisms for loss of joint tightness for these items are IC and ISR. Fatigue and Wear, which can also lead to cracking, are secondary aging degradation mechanisms after significant stress relaxation and loss of preload has occurred due to IC/ISR. Bolt stress relaxation cannot be readily inspected by NDE. Only bolt cracking is inspected by UT inspection in this table. The effect of loss of closure integrity on the functionality will be addressed by analysis of the core barrel assembly.

Change h:

As noted in the meeting between the NRC and MRP/Industry on 10/14/2010, AREVA is working with its owners to support implementation of MRP-227. As part of this effort, it was determined that the words—no examination requirements—were omitted from the "Examination Method" column for the external baffle-to-baffle bolts locking devices, including locking welds and for the core barrel-to-former bolts locking devices, including locking welds. This component is listed as part of the Core Barrel Assembly in Table 4-4 of MRP-227, Rev. 0 as shown below.

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method (Note 1)	Examination Coverage
Core Barrel Assembly Locking devices, including locking welds, for the external baffle-to-baffle bolts and core barrel-to-former bolts	All plants	Cracking (IASCC, IE)	Locking devices, including locking welds, of baffle-to-former bolts or internal baffle-to-baffle bolts	Justify by evaluation or by replacement.	Inaccessible. See Figure 4-2.

In order to reflect this, MRP proposes that the following wording to be inserted into the "Examination Method" column of Table 4-4 of MRP-227-A.

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method (Note 1)	Examination Coverage
Core Barrel Assembly Locking devices, including locking welds, for the external baffle-to-baffle bolts and core barrel-to-former bolts	All plants	Cracking (IASCC, IE)	Locking devices, including locking welds, of baffle-to-former bolts or internal baffle-to-baffle bolts	No examination requirements. Justify by evaluation or by replacement.	Inaccessible. See Figure 4-2.

Change i:

As noted in the meeting between the NRC and MRP/Industry on 10/14/2010, AREVA is working with its owners to support implementation of MRP-227. As part of this effort, it was determined that the "effect (mechanism)" column only identifies the "effect (mechanism)" for the lower grid shock pad bolts and does not clearly describe the "effect (mechanism)" for the associated locking devices. Also, the associated locking devices were omitted from the "Item" and "Primary Link" columns. In addition, the locking devices were omitted from the "Examination Method" and "Examination Coverage" columns. This component is listed as part of the Core Barrel Assembly in Table 4-4 of MRP-227, Rev. 0 as shown below.

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method (Note 1)	Examination Coverage
Lower Grid Assembly Lower grid shock pad bolts	TMI-1	Cracking (SCC)	UCB and LCB bolts	Volumetric examination (UT)	100% of accessible bolts See Figure 4-4

In order to reflect these above omissions, MRP proposes that the additional wording be inserted into Table 4-4 of MRP-227-A.

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method (Note 1)	Examination Coverage
Lower Grid Assembly Lower grid shock pad bolts and their locking devices	TMI-1	Bolt: Cracking (SCC) Locking Devices: Loss of material, damaged, distorted, or missing locking devices (Wear or Fatigue damage by failed bolts)	UCB and LCB bolts and their locking devices	Bolt: Volumetric examination (UT). Locking Devices: Visual (VT-3) examination.	100% of accessible bolts and locking devices. See Figure 4-4.

Change j:

As noted in the meeting between the NRC and MRP/Industry on 10/14/2010, AREVA is working with its owners to support implementation of MRP-227. As part of this effort, it was determined that the "effect (mechanism)" column only identifies the "effect (mechanism)" for the LTS bolts and flow distributor bolts and does not clearly describe the "effect (mechanism)" for the associated locking devices. Also, the associated locking devices were omitted from the "Item" and "Primary Link" columns. In addition, the locking devices were omitted from the "Examination Method" and "Examination Coverage" columns. This component is listed as part of the Core Barrel Assembly in Table 4-4 of MRP-227, Rev. 0 as shown below.

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method (Note 1)	Examination Coverage
Lower Grid Assembly Lower thermal shield bolts (LTS)	All plants	Cracking (SCC)	UCB and LCB bolts	Volumetric examination (UT)	100% of accessible bolts
Flow Distributor Assembly Flow distributor bolts (FD)					See Figure 4-8

In order to reflect these above omissions, MRP proposes that the additional wording be inserted into Table 4-4 of MRP-227-A.

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method (Note 1)	Examination Coverage
Lower Grid Assembly Lower thermal shield bolts (LTS) and their locking devices	All plants	Bolt: Cracking (SCC) Locking Devices: Loss of material, damaged, distorted, or missing locking devices (Wear or Fatigue damage by failed bolts)	UCB and LCB bolts and their locking devices	Bolt: Volumetric examination (UT). Locking Devices: Visual (VT-3) examination.	100% of accessible bolts and locking devices.
Flow Distributor Assembly Flow distributor bolts (FD) and their locking devices					See Figure 4-8.

Change k:

As noted in the meeting between the NRC and MRP/Industry on 10/14/2010, AREVA is working with its owners to support implementation of MRP-227. As part of a records review and accessibility evaluation, it was determined that the "CSS vent valve disc shaft or hinge pin" was inaccessible. This component is listed as part of the Core Support Shield Assembly in Table 5-1 of MRP-227, Rev. 0 as shown below.

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Support Shield Assembly CSS vent valve top retaining ring CSS vent valve bottom retaining ring CSS vent valve disc shaft or hinge pin	All plants	Visual (VT-3) examination. The specific relevant condition is evidence of damaged or fractured material, and missing items	None	N/A	N/A

In order to reflect the actual generic condition and to clarify the requirements, MRP proposes that the following two rows be inserted into MRP-227-A and the existing row (shown above) be deleted.

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Support Shield Assembly CSS vent valve top retaining ring CSS vent valve bottom retaining ring	All plants	Visual (VT-3) examination. The specific relevant condition is evidence of damaged or fractured material, and missing items	None	N/A.	N/A.
Core Support Shield Assembly CSS vent valve disc shaft or hinge pin	All plants	Inaccessible. Justify by evaluation or replacement.	None	N/A.	N/A.

Change 1:

As noted in the meeting between the NRC and MRP/Industry on 10/14/2010, AREVA is working with its owners to support implementation of MRP-227. As part of this effort, it was determined that associated locking devices were omitted from the "Expansion Link(s)" column for each of the expansion link bolts. This component is listed as part of the Core Support Shield Assembly in Table 5-1 of MRP-227, Rev. 0 as shown below.

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Support Shield Assembly Upper core barrel (UCB) bolts and their locking devices	All plants	1) Volumetric (UT) examination of the UCB bolts. The examination acceptance criteria for the UT of the UCB bolts shall be established as part of the examination technical justification. 2) Visual (VT-3) examination of the UCB bolt locking devices. The specific relevant condition for the VT-3 of the UCB bolt locking devices is evidence of broken or missing bolt locking devices.	LCB (Note 2) UTS, LTS, and FD bolts SSHT bolts (CR-3 and DB only) Lower grid shock pad bolts (TMI-1 only)	1) Confirmed unacceptable indications exceeding 10% of the UCB bolts shall require that the UT examination be expanded by the completion of the next refueling outage to include: For all plants 100% of the accessible UTS bolts, 100% of the accessible LTS bolts, 100% of the accessible FD bolts, Additionally for TMI-1 UT examination to include 100% of the accessible lower grid shock pad bolts, Additionally for CR-3 and DB UT examination to include 100% of the accessible SSHT bolts. 2) Confirmed evidence of relevant conditions exceeding 10% of the UCB bolt locking devices shall require that the VT-3 examination be expanded by the completion of the next refueling outage to include: For all plants 100% of the accessible UTS, LTS, and FD bolt locking devices, Additionally for TMI-1 100% of the accessible lower grid shock pad bolt locking devices,	1) The examination acceptance criteria for the UT of the expansion bolting shall be established as part of the examination technical justification. 2) The specific relevant condition for the expansion of the VT-3 locking devices is evidence of broken or missing bolt locking devices.

In order to reflect these above omissions, MRP proposes that the additional wording be inserted into Table 5-1 of MRP-227-A.

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Support Shield Assembly Upper core barrel (UCB) bolts and their locking devices	All plants	1) Volumetric (UT) examination of the UCB bolts. The examination acceptance criteria for the UT of the UCB bolts shall be established as part of the examination technical justification. 2) Visual (VT-3) examination of the UCB bolt locking devices. The specific relevant condition for the VT-3 of the UCB bolt locking devices is evidence of broken or missing bolt locking devices.	LCB and their locking devices (Note 2) UTS, LTS, and FD bolts and their locking devices SSHT bolts and their locking devices (CR-3 and DB only) Lower grid shock pad bolts and their locking devices (TMI-1 only)	1) Confirmed unacceptable indications exceeding 10% of the UCB bolts shall require that the UT examination be expanded by the completion of the next refueling outage to include: For all plants 100% of the accessible UTS bolts, 100% of the accessible LTS bolts, 100% of the accessible FD bolts, Additionally for TMI-1 UT examination to include 100% of the accessible lower grid shock pad bolts, Additionally for CR-3 and DB UT examination to include 100% of the accessible SSHT bolts. 2) Confirmed evidence of relevant conditions exceeding 10% of the UCB bolt locking devices shall require that the VT-3 examination be expanded by the completion of the next refueling outage to include: For all plants 100% of the accessible UTS, LTS, and FD bolt locking devices, Additionally for TMI-1 100% of the accessible lower grid shock pad bolt locking devices, Additionally for CR-3 and DB	1) The examination acceptance criteria for the UT of the expansion bolting shall be established as part of the examination technical justification. 2) The specific relevant condition for the expansion of the VT-3 locking devices is evidence of broken or missing bolt locking devices.

Change m:

As noted in the meeting between the NRC and MRP/Industry on 10/14/2010, AREVA is working with its owners to support implementation of MRP-227. As part of this effort, it was determined that associated locking devices were omitted from the "Expansion Link(s)" column for each of the expansion link bolts. This component is listed as part of the Core Barrel Assembly in Table 5-1 of MRP-227, Rev. 0 as shown below.

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Barrel Assembly Lower core barrel (LCB) bolts and their locking devices	All plants	1) Volumetric (UT) examination of the LCB bolts. The examination acceptance criteria for the UT of the LCB bolts shall be established as part of the examination technical justification. 2) Visual (VT-3) examination of the LCB bolt locking devices. The specific relevant condition for the VT-3 of the LCB bolt locking devices is evidence of broken or missing bolt locking devices.	UTS, LTS, and FD bolts SSHT bolts (CR-3 and DB only) Lower grid shock pad bolts (TMI-1 only)	1) Confirmed unacceptable indications exceeding 10% of the LCB bolts shall require that the UT examination be expanded by the completion of the next refueling outage to include: For all plants 100% of the accessible UTS bolts, 100% of the accessible LTS bolts, 100% of the accessible FD bolts, Additionally for TMI-1 100% of the accessible lower grid shock pad bolts, Additionally for CR-3 and DB 100% of the accessible SSHT bolts by the completion of the next refueling outage. 2) Confirmed evidence of relevant conditions exceeding 10% of the LCB bolt locking devices shall require that the VT-3 examination be expanded by the completion of the next refueling outage to include: For all plants 100% of the accessible UTS	1) The examination acceptance criteria for the UT of the expansion bolting shall be established as part of the examination technical justification. 2) The specific relevant condition for the expansion of the VT-3 of the locking devices is evidence of broken or missing bolt locking devices.

In order to reflect these above omissions, MRP proposes that the additional wording be inserted into Table 5-1 of MRP-227-A.

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Barrel Assembly Lower core barrel (LCB) bolts and their locking devices	All plants	1) Volumetric (UT) examination of the LCB bolts. The examination acceptance criteria for the UT of the LCB bolts shall be established as part of the examination technical justification. 2) Visual (VT-3) examination of the LCB bolt locking devices. The specific relevant condition for the VT-3 of the LCB bolt locking devices is evidence of broken or missing bolt locking devices.	UTS, LTS, and FD bolts and their locking devices SSHT bolts and their locking devices (CR-3 and DB only) Lower grid shock pad bolts and their locking devices (TMI-1 only)	1) Confirmed unacceptable indications exceeding 10% of the LCB bolts shall require that the UT examination be expanded by the completion of the next refueling outage to include: For all plants 100% of the accessible UTS bolts, 100% of the accessible LTS bolts, 100% of the accessible FD bolts, Additionally for TMI-1 100% of the accessible lower grid shock pad bolts, Additionally for CR-3 and DB 100% of the accessible SSHT bolts by the completion of the next refueling outage. 2) Confirmed evidence of relevant conditions exceeding 10% of the LCB bolt locking devices shall require that the VT-3 examination be expanded by the completion of the next refueling outage to include: For all plants 100% of the accessible UTS	1) The examination acceptance criteria for the UT of the expansion bolting shall be established as part of the examination technical justification. 2) The specific relevant condition for the expansion of the VT-3 of the locking devices is evidence of broken or missing bolt locking devices.

Change n:

As noted in the meeting between the NRC and MRP/Industry on 10/14/2010, AREVA is working with its owners to support implementation of MRP-227. As part of this effort, it was determined that an incorrect wording ("former plate" in lieu of "baffle plate") was erroneously used in the "Expansion Criteria" column for the Core Barrel Assembly Baffle-to-former bolts component. This component is listed as part of the Core Barrel Assembly in Table 5-1 of MRP-227, Rev. 0 as shown below.

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Barrel Assembly Baffle-to-former bolts	All plants	Baseline volumetric (UT) examination of the baffle-to-former bolts. The examination acceptance criteria for the UT of the baffle-to-former bolts shall be established as part of the examination technical justification.	Baffle-to-baffle bolts, Core barrel-to-former bolts	Confirmed unacceptable indications in greater than or equal to 5% (or 43) of the baffle-to-former bolts, provided that none of the unacceptable bolts are on former elevations 3, 4, and 5, or greater than 25% of the bolts on a single former plate, shall require an evaluation of the internal baffle-to-baffle bolts for the purpose of determining whether to examine or replace the internal baffle-to-baffle bolts. The evaluation may include external baffle-to-baffle bolts and core barrel-to-former bolts for the purpose of determining whether to replace them.	N/A

In order to reflect correction of this wording, MRP proposes that the modified wording be inserted into Table 5-1 of MRP-227-A.

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Barrel Assembly Baffle-to-former bolts	All plants	Baseline volumetric (UT) examination of the baffle-to-former bolts. The examination acceptance criteria for the UT of the baffle-to-former bolts shall be established as part of the examination technical justification.	Baffle-to-baffle bolts, Core barrel-to-former bolts	Confirmed unacceptable indications in greater than or equal to 5% (or 43) of the baffle-to-former bolts, provided that none of the unacceptable bolts are on former elevations 3, 4, and 5, or greater than 25% of the bolts on a single baffle plate, shall require an evaluation of the internal baffle-to-baffle bolts for the purpose of determining whether to examine or replace the internal baffle-to-baffle bolts. The evaluation may include external baffle-to-baffle bolts and core barrel-to-former bolts for the purpose of determining whether to replace them.	N/A

5) The MRP proposes the following changes to the CE and Westinghouse tables in MRP-227-Rev. 0. Changes are indicated in track changes as well as bar on the left side of each row with changes. Note that these tables do not include all of the proposed changes to the CE and Westinghouse Tables (see point 3). A set of tables with all the changes combined could be provided to the NRC later if necessary.

   a) The MRP proposes to use the additions in the Effect (Mechanism) column in Tables 4-2, 4-3, 4-5, 4-6, 4-8 and 4-9 as shown in the tables below.

   b) The MRP proposes to clarify some of the 4-2 CE table entries for TLAA/fatigue analysis by replacing the words "plant-specific fatigue analysis" with the words "evaluation to determine the potential location and extent of fatigue cracking" as shown in the tables below.

   c) The MRP proposes to replace the title in column 4 "Primary Link" with "Reference" for Tables 4-8 and 4-9 as shown in the tables below. The MRP proposes to make the Westinghouse "Remaining core barrel welds" consistent between Tables 3-3, 4-3, 4-6 and 5-3 as shown in the tables below.

   d) The MRP proposes to delete the sentence "Replacement of 304 springs by 403 springs is required when the spring stiffness is determined to relax beyond design tolerance" in the Westinghouse Plants Primary Components Table (Table 4-1) for the Alignment and Interfacing Components Internals hold down spring item.

   e) The MRP proposes to delete the text "or as supported by plant-specific justification" for the core-shroud bolts item in CE Table 4-2 and the baffle-former bolts item in Westinghouse Table 4-3.

Table 4-2
CE Plants Primary Components

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Core Shroud Assembly (Bolted) Core shroud bolts	Bolted plant designs	Cracking (IASCC, Fatigue) Aging Management (IE and ISR)	Core support column bolts, Barrel-shroud bolts	Baseline volumetric (UT) examination between 25 and 35 EFPY, with subsequent examination after 10 to 15 additional EFPY to confirm stability of bolting pattern. Re-examination for high-leakage core designs requires continuing inspections on a ten-year interval.	100% of accessible bolts~~, or as supported by plant-specific justification.~~. Heads are accessible from the core side. UT accessibility may be affected by complexity of head and locking device designs. See Figure 4-24.
Core Shroud Assembly (Welded) Core shroud plate-former plate weld	Plant designs with core shrouds assembled in two vertical sections	Cracking (IASCC) Aging Management (IE)	Remaining axial welds	Enhanced visual (EVT-1) examination no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	Axial and horizontal weld seams at the core shroud re-entrant corners as visible from the core side of the shroud, within six inches of central flange and horizontal stiffeners. See Figures 4-12 and 4-14.
Core Shroud Assembly (Welded) Shroud plates	Plant designs with core shrouds assembled with full-height shroud plates	Cracking (IASCC) Aging Management (IE)	Remaining axial welds, ribs and rings	Enhanced visual (EVT-1) examination no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	Axial weld seams at the core shroud re-entrant corners, at the core mid-plane (± three feet in height) as visible from the core side of the shroud. See Figure 4-13.

Table 4-2
CE Plants Primary Components

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Core Shroud Assembly (Bolted) Assembly	Bolted plant designs	Distortion (Void Swelling) including: • Abnormal interaction with fuel assemblies • Gaps along high fluence shroud plate joints • Vertical displacement of shroud plates near high fluence joint Aging Management (IE)	None	Visual (VT-3) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	Core side surfaces as indicated. See Figures 4-25 and 4-26.
Core Shroud Assembly (Welded) Assembly	Plant designs with core shrouds assembled in two vertical sections	Distortion (Void Swelling), as evidenced by separation between the upper and lower core shroud segments Aging Management (IE)	None	Visual (VT-1) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	If a gap exists, make three to five measurements of gap opening from the core side at the core shroud re-entrant corners. Then, evaluate the swelling on a plant-specific basis to determine frequency and method for additional examinations. See Figures 4-12 and 4-14.
Core Support Barrel Assembly Upper (core support barrel) flange weld	All plants	Cracking (SCC)	Remaining core barrel assembly welds, core support column welds	Enhanced visual (EVT-1) examination no later than two refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	100% of the accessible surfaces of the upper flange weld. See Figure 4-15.

Table 4-2
CE Plants Primary Components

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Core Support Barrel Assembly Lower flange weld	All plants	Cracking (Fatigue)	None	If fatigue life cannot be demonstrated by time-limited aging analysis (TLAA), enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examination on a ten-year interval.	Examination coverage to be defined by ~~plant-specific fatigue analysis~~evaluation to determine the potential location and extent of fatigue cracking. See Figure 4-15.
Lower Support Structure Core support plate	All plants with a core support plate	Cracking (Fatigue) Aging Management (IE)	None	If fatigue life cannot be demonstrated by time-limited aging analysis (TLAA), enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examination on a ten-year interval.	Examination coverage to be defined by ~~plant-specific fatigue analysis~~evaluation to determine the potential location and extent of fatigue cracking. See Figure 4-16.
Upper Internals Assembly Fuel alignment plate	All plants with core shrouds assembled with full-height shroud plates	Cracking (Fatigue)	None	If fatigue life cannot be demonstrated by time-limited aging analysis (TLAA), enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examination on a ten-year interval.	Examination coverage to be defined by ~~plant-specific fatigue analysis~~evaluation to determine the potential location and extent of fatigue cracking. See Figure 4-17.

Table 4-2
CE Plants Primary Components

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Control Element Assembly Instrument guide tubes	All plants with instrument guide tubes in the CEA shroud assembly	Cracking (SCC, Fatigue) that results in missing supports or separation at the welded joint between the tubes and supports	Remaining instrument guide tubes within the CEA shroud assemblies	Visual (VT-3) examination, no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examination on a ten-year interval. Plant-specific component integrity assessments may be required if degradation is detected and remedial action is needed.	100% of tubes in peripheral CEA shroud assemblies (i.e., those adjacent to the perimeter of the fuel alignment plate). See Figure 4-18.
Lower Support Structure Deep beams	All plants with core shrouds assembled with full-height shroud plates	Cracking (Fatigue) that results in a detectable surface-breaking indication in the welds or beams Aging Management (IE)	None	Enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examination on a ten-year interval, if adequacy of remaining fatigue life cannot be demonstrated.	Examine beam-to-beam welds, in the axial elevation from the beam top surface to 4 inches below. See Figure 4-19.

Note: 1. Examination acceptance criteria and expansion criteria for the CE components are in Table 5-2.

Table 4-3
Westinghouse Plants Primary Components

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Control Rod Guide Tube Assembly Guide plates (cards)	All plants	Loss of Material (Wear)	None	Visual (VT-3) examination no later than 2 refueling outages from the beginning of the license renewal period, and no earlier than two refueling outages prior to the start of the license renewal period. Subsequent examinations are required on a ten-year interval.	20% examination of the number of CRGT assemblies, with all guide cards within each selected CRGT assembly examined. See Figure 4-20.
Control Rod Guide Tube Assembly Lower flange welds	All plants	Cracking (SCC, Fatigue) Aging Management (IE and TE)	Bottom-mounted instrumentation (BMI) column bodies, Lower support column bodies (cast)	Enhanced visual (EVT-1) examination to determine the presence of crack-like surface flaws in flange welds no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	100% of outer (accessible) CRGT lower flange weld surfaces and adjacent base metal. See Figure 4-21.
Core Barrel Assembly Upper core barrel flange weld	All plants	Cracking (SCC)	Remaining core barrel welds, Lower support column bodies (non cast)	Periodic enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal. See Figure 4-22.

Table 4-3
Westinghouse Plants Primary Components

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Baffle-Former Assembly Baffle-edge bolts	All plants with baffle-edge bolts	Cracking (IASCC, Fatigue) that results in • Lost or broken locking devices • Failed or missing bolts • Protrusion of bolt heads Aging Management (IE and ISR)	None	Visual (VT-3) examination, with baseline examination between 20 and 40 EFPY and subsequent examinations on a ten-year interval.	Bolts and locking devices on high fluence seams. 100% of components accessible from core side. See Figure 4-23.
Baffle-Former Assembly Baffle-former bolts	All plants	Cracking (IASCC, Fatigue) Aging Management (IE and ISR)	Lower support column bolts, Barrel-former bolts	Baseline volumetric (UT) examination between 25 and 35 EFPY, with subsequent examination after 10 to 15 additional EFPY to confirm stability of bolting pattern. Re-examination for high-leakage core designs requires continuing examinations on a ten-year interval.	100% of accessible bolts. ~~or as supported by plant-specific justification.~~ Heads accessible from the core side. UT accessibility may be affected by complexity of head and locking device designs. See Figures 4-23 and 4-24.

Table 4-3
Westinghouse Plants Primary Components

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Baffle-Former Assembly Assembly (Includes: ~~bB~~Baffle ~~pP~~plates, ~~bB~~Baffle-~~E~~edge ~~B~~bolts~~. Also~~, and indirect effects of void swelling in ~~f~~Former plates).	All plants	Distortion (Void Swelling), or Cracking (IASCC) that results in: • Abnormal interaction with fuel assemblies • Gaps along high fluence baffle joint • Vertical displacement of baffle plates near high fluence joint • Broken or damaged edge bolt locking systems along high fluence baffle joints	None	Visual (VT-3) examination to check for evidence of distortion, with baseline examination between 20 and 40 EFPY and subsequent examinations on a ten-year interval.	Core side surface as indicated. See Figures 4-24, 4-25, 4-26 and 4-27.
Alignment and Interfacing Components Internals hold down spring	All plants with 304 stainless steel hold down springs	Distortion (Loss of Load) Note: This mechanism was not strictly identified in the original list of age-related degradation mechanisms [7].	None	Direct measurement of spring height within three cycles of the beginning of the license renewal period. If the first set of measurements is not sufficient to determine life, spring height measurements must be taken during the next two outages, in order to extrapolate the expected spring height to 60 years.	Measurements should be taken at several points around the circumference of the spring, with a statistically adequate number of measurements at each point to minimize uncertainty. ~~Replacement of 304 springs by 403 springs is required when the spring stiffness is determined to relax beyond design tolerance.~~ See Figure 4-28.

Table 4-3
Westinghouse Plants Primary Components

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Thermal Shield Assembly Thermal shield flexures	All plants with thermal shields	Cracking (Fatigue) or Loss of Material (Wear) that results in thermal shield flexures excessive wear, fracture, or complete separation	None	Visual (VT-3) no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	100% of thermal shield flexures. See Figures 4-29 and 4-36.

Note:
1. Examination acceptance criteria and expansion criteria for the Westinghouse components are in Table 5-3.

Table 4-5
CE Plants Expansion Components

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method (Note 1)	Examination Coverage
Core Shroud Assembly (Bolted) Barrel-shroud bolts	Bolted plant designs	Cracking (IASCC, Fatigue) Aging Management (IE and ISR)	Core shroud bolts	Volumetric (UT) examination, with initial and subsequent examination frequencies dependent on the results of core shroud bolt examinations.	100% (or as supported by plant-specific justification) of barrel-shroud and guide lug insert bolts with neutron fluence exposures > 3 displacements per atom (dpa). See Westinghouse design Figure 4-23.
Core Support Barrel Assembly Lower core barrel flange	All plants	Cracking (SCC, Fatigue)	Upper (core support barrel) flange weld	Enhanced visual (EVT-1) examination, with initial and subsequent examinations dependent on the results of the upper (core support barrel) flange weld examinations.	100% of accessible welds and adjacent base metal. See Figure 4-15.
Core Support Barrel Assembly Remaining core barrel assembly welds	All plants	Cracking (SCC) Aging Management (IE)	Upper (core support barrel) flange weld	Enhanced visual (EVT-1) examination, with initial and subsequent examinations dependent on the results of core barrel assembly upper flange weld examinations.	100% of one side of the accessible weld and adjacent base metal surfaces for the weld with the highest calculated operating stress. See Figure 4-15.
Lower Support Structure Core support column welds	All plants except those with core shrouds assembled with full-height shroud plates	Cracking (SCC, IASCC, Fatigue) including damaged or fractured material Aging Management (IE)	Upper (core support barrel) flange weld	Visual (VT-3) examination, with initial and subsequent examinations based on plant evaluation of SCC susceptibility and demonstration of remaining fatigue life.	Examination coverage determined by plant-specific analysis. See Figures 4-16 and 4-31.

Table 4-5
CE Plants Expansion Components

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method (Note 1)	Examination Coverage
Core Shroud Assembly (Bolted) Core support column bolts	Bolted plant designs	Cracking (IASCC, Fatigue) Aging Management (IE)	Core shroud bolts	Ultrasonic (UT) examination, with initial and subsequent examination frequencies dependent on the results of core shroud bolt examinations.	100% (or as supported by plant-specific analysis) of core support column bolts with neutron fluence exposures >3 dpa. See Figures 4-16 and 4-33.
Core Shroud Assembly (Welded) Remaining axial welds	Plant designs with core shrouds assembled in two vertical sections	Cracking (IASCC)	Core shroud plate-former plate weld	Enhanced visual (EVT-1) examination, with initial and subsequent examination frequencies dependent on the results of the core shroud weld examinations.	Axial weld seams other than the core shroud re-entrant corner welds at the core mid-plane. See Figure 4-12.
Core Shroud Assembly (Welded) Remaining axial welds Ribs and rings	Plant designs with core shrouds assembled with full-height shroud plates	Cracking (IASCC) Aging Management (IE)	Shroud plates of welded core shroud assemblies	Enhanced visual (EVT-1) examination, with initial and subsequent examination frequencies dependent on the results of the core shroud weld examinations.	Axial weld seams other than core shroud re-entrant corner welds at the core mid-plane, plus ribs and rings. See Figure 4-13.
Control Element Assembly Remaining instrument guide tubes	All plants with instrument guide tubes in the CEA shroud assembly	Cracking (SCC, Fatigue) that results in missing supports or separation at the welded joint between the tubes and supports.	Peripheral instrument guide tubes within the CEA shroud assemblies	Visual (VT-3) examination, with initial and subsequent examinations dependent on the results of the instrument guide tubes examinations.	100% of tubes in CEA shroud assemblies. See Figure 4-18.

Note: 1. Examination acceptance criteria and expansion criteria for the CE components are in Table 5-2.

Table 4-6
Westinghouse Plants Expansion Components

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method (Note 1)	Examination Coverage
Core Barrel Assembly Barrel-former bolts	All plants	Cracking (IASCC, Fatigue) Aging Management (IE, Void Swelling and ISR)	Baffle-former bolts	Volumetric (UT) examination, with initial and subsequent examinations dependent upon results of baffle-former bolt examinations.	100% of accessible bolts. Accessibility may be limited by presence of thermal shields or neutron pads. See Figure 4-23.
Lower Support Assembly Lower support column bolts	All plants	Cracking (IASCC, Fatigue) Aging Management (IE and ISR)	Baffle-former bolts	Volumetric (UT) examination, with initial and subsequent examinations dependent on results of baffle-former bolt examinations.	100% of accessible bolts or as supported by plant-specific justification. See Figures 4-32 and 4-33.
Core Barrel Assembly Remaining Welds (Core barrel flanges, core barrel outlet nozzles)~~, lower core barrel flange weld~~	All plants	Cracking (SCC, Fatigue) Aging Management (IE of lower sections)	Upper core barrel flange weld	Enhanced visual (EVT-1) examination, with initial examination and re-examination frequency dependent on the examination results for upper core barrel flange.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal. See Figure 4-22
Lower Support Assembly Lower support column bodies (non cast)	All plants	Cracking (IASCC) Aging Management (IE)	Upper core barrel flange weld	Enhanced visual (EVT-1) examination, with initial examination and re-examination frequency dependent on the examination results for upper core barrel flange weld.	100% of accessible surfaces. See Figure 4-34.

Table 4-6
Westinghouse Plants Expansion Components

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method (Note 1)	Examination Coverage
Lower Support Assembly Lower support column bodies (cast)	All plants	Cracking (IASCC) including the detection of fractured support columns Aging Management (IE)	Control rod guide tube (CRGT) lower flanges	Visual (EVT-1) examination.	100% of accessible support columns. See Figure 4-34.
Bottom-Mounted Instrumentation System Bottom-Mounted Instrumentation (BMI) column bodies	All plants	Cracking (Fatigue) including the detection of completely fractured column bodies Aging Management (IE)	Control rod guide tube (CRGT) lower flanges	Visual (VT-3) examination of BMI column bodies as indicated by difficulty of insertion/withdrawal of flux thimbles. Flux thimble insertion/withdrawal to be monitored at each inspection interval.	100% of BMI column bodies for which difficulty is detected during flux thimble insertion/withdrawal See Figures 4-35.

Note:

1. Examination acceptance criteria and expansion criteria for the Westinghouse components are in Table 5-3.

Table 4-8
CE Plants Existing Programs Components

Item	Applicability	Effect (Mechanism)	~~Primary Link~~Reference	Examination Method	Examination Coverage
Core Shroud Assembly Guide lugs Guide lug inserts and bolts	All plants	Loss of material (Wear) Aging Management (ISR)	ASME Code Section XI	Visual (VT-3) examination, general condition examination for detection of excessive or asymmetrical wear.	First 10-year ISI after 40 years of operation, and at each subsequent inspection interval.
Lower Support Structure Fuel alignment pins	All plants with core shrouds assembled with full-height shroud plates	Cracking (SCC, IASCC, Fatigue) Aging Management (IE and ISR)	ASME Code Section XI	Visual (VT-3) examination to detect severed fuel alignment pins, missing locking tabs, or excessive wear on the fuel alignment pin nose or flange.	Accessible surfaces at specified frequency.
Lower Support Structure Fuel alignment pins	All plants with core shroud assembled in two vertical sections	Loss of material (Wear) Aging Management (IE and ISR)	ASME Code Section XI	Visual (VT-3) examination.	Accessible surfaces at specified frequency.
Core Barrel Assembly Upper flange	All plants	Loss of material (Wear)	ASME Code Section XI	Visual (VT-3) examination.	Area of the upper flange potentially susceptible to wear.

Table 4-9
Westinghouse Plants Existing Programs Components

Item	Applicability	Effect (Mechanism)	~~Expansion Link~~Reference	Examination Method	Examination Coverage
Core Barrel Assembly Core barrel flange	All plants	Loss of material (Wear)	ASME Code Section XI	Visual (VT-3) examination to determine general condition for excessive wear.	All accessible surfaces at specified frequency.
Upper Internals Assembly Upper support ring or skirt	All plants	Cracking (SCC, Fatigue)	ASME Code Section XI	Visual (VT-3) examination.	All accessible surfaces at specified frequency.
Lower Internals Assembly Lower core plate XL lower core plate (Note 1)	All plants	Cracking (IASCC, Fatigue) Aging Management (IE)	ASME Code Section XI	Visual (VT-3) examination of the lower core plates to detect evidence of distortion and/or loss of bolt integrity.	All accessible surfaces at specified frequency.
Lower Internals Assembly Lower core plate XL lower core plate (Note 1)	All plants	Loss of material (Wear)	ASME Code Section XI	Visual (VT-3) examination.	All accessible surfaces at specified frequency.
Bottom Mounted Instrumentation System Flux thimble tubes	All plants	Loss of material (Wear)	NUREG-1801 Rev. 1	Surface (ET) examination.	Eddy current surface examination as defined in plant response to IEB 88-09.
Alignment and Interfacing Components Clevis insert bolts	All plants	Loss of material (Wear) (Note 2)	ASME Code Section XI	Visual (VT-3) examination.	All accessible surfaces at specified frequency.
Alignment and Interfacing Components Upper core plate alignment pins	All plants	Loss of material (Wear)	ASME Code Section XI	Visual (VT-3) examination.	All accessible surfaces at specified frequency.

Notes:
1. XL = "Extra Long" referring to Westinghouse plants with 14-foot cores.
2. Bolt was screened in because of stress relaxation and associated cracking; however, wear of the clevis/insert is the issue.

6) The MRP proposes to add the following reference to Section 8 of MRP-227-Rev. 0: "[26]. WCAP-17096-NP, "Reactor Internals Acceptance Criteria Methodology and Data Requirements - Revision 2", December 2009."

7) The MRP proposes to replace the current Appendix A in MRP-227-Rev. 0 called Aging Management Program Attributes by the EPRI DRAFT Input (12-01-09): New Appendix A to MRP-227 A called Operating Experience Summary provided in letter MRP 2009-091 (Subject: Transmittal of Initial Draft Material to Support NRC Update of NUREG 1801, "Generic Aging Lessons Learned Report" (GALL)) sent to the NRC in December 2009.

8) The MRP proposes to replace the words in last paragraph Section 7.1 of MRP-227 with clarifying words about NEI-03-08. Specifically the words "Addendum D to NEI 03-08 [1]" with the following: "Addendum E to NEI 03-08, Revision 2". Reference 1 will also be updated to reflect NEI 03-08, Revision 2, January 2010.

Appendix B

MRP-227 Roadmap

MRP-227 Roadmap
October 29, 2010

The following road map is intended to provide information to NRC staff that will facilitate their review of MRP-227. The goal is not to tell the technical story in a different fashion, but rather to provide an overview of the steps involved in development of MRP-227 and point the staff to the appropriate supporting documents. In preparing this roadmap, no new information has been provided. Everything noted in this roadmap has been excerpted from other references previously provided to the NRC staff as part of the MRP-227 review and RAI process.

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

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 support reactor internals aging management for the current license period as well.

It is important to recognize that this effort relied on the previous work in MRP-205 (Issue Management Tables). These tables identified all safety significant issues for all PWR primary loop and internals components. Further, only two components were identified during the initial screening (step 1) that had any safety consequences that were dispositioned in the development of MRP-227; as explained in this roadmap.

The guidelines are applicable to nuclear steam supply system (NSSS) vendor Babcock & Wilcox-designed (B&W), Combustion Engineering-designed (CE) and Westinghouse-designed (W) PWR internals. The guidelines are based on a broad set of assumptions about nuclear unit operation, which encompass the range of current unit conditions for the U.S. fleet of PWRs. The aging management strategy reports, MRP-231 for B&W and MRP-232 for CE and W, provide the basis for these guidelines. The functional evaluations, including the screening and the Failure Modes, Effects and Criticality Analysis (FMECA), that support the guidelines were based on representative B&W, W and CE PWR reactor vessel internals configurations, existing analyses, inspections, and operational histories, which were generally conservative, but not necessarily bounding in every parameter.

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 unit-specific licensing inservice inspection requirements. The guidelines do not replace the current licensing basis for the current and extended license periods, which have been reviewed and approved by the US NRC on a plant-specific basis based on NUREG-1800 and NUREG-1801.

The goal is to ensure the long-term safety, integrity, and reliability of PWR internals using proven and familiar methods for inspection, monitoring, surveillance, and reporting.

MRP-227 Roadmap
October 29, 2010

An experienced team consisting of utility, NSSS vendor 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 process used to develop the MRP-227 recommendations may be described in terms of the following sequence of steps:

Step 1 – Identify PWR internals components, materials, and environments
Step 2 – Identify degradation screening criteria
Step 3 – Characterize components and screen for degradation (A, non-A)
Step 4 – FMECA Review
Step 5 – Severity categorization (A, B, C)
Step 6 – Engineering Evaluation and Assessment[1]
Step 7 – Categorize for Inspection (Primary, Expansion, Existing, No Additional Measures) and Aging Management Strategy
Step 8 – Preparation of MRP-227 I&E Guidelines

The processing of the reactor internals components through these eight steps is outlined in the following paragraphs. The screening and categorization processes for B&W components is are contained described in MRP-189 Rev. 1, MRP-190, and MRP-231. The screening and categorization processes for0 and the W and CE internals are described in MRP-191 and MRP-232.

In addition to the documents specifically focused on PWR reactor internals, two other resources were utilized – the Materials Degradation Matrix (MDM) and the PWR Issue Management Tables (IMTs) that are compiled in MRP-205, rRev. 1. The MDM was first issued in 2004. It documents all known relevant/plausible degradation mechanisms and materials, including welds, in the primary loop and reactor internals for BWRs and PWRsS. This document was developed with the support of domestic and international experts from NSSS vendors, national laboratories, utilities and consultants. (It is worth noting that NRC conducted a similar activity that is documented in their Expert Panel Report on Proactive Materials Degradation Assessment NUREG/CR-6923. It reached essentially the same conclusions.) The PWR IMTs used the information from the MDM and assessed, at a component level the consequences of failure, as well as inspection, mitigation and repair technology associated with that component. The MDM and IMTs are maintained as "living documents" and updated periodically.

Key to the development of MRP-205 was the extensive efforts by the NSSS vendors, key utility personnel and supporting experts to identify the failure consequences at a component level. This work is described in MRP-157 for B&W plants and in MRP-156 for W and CE plants. These documents were used extensively in the overall development of MRP-227.

[1] Step 6 has previously been identified as a "Functionality Evaluation" or "Functionality Assessment" in each of the reference documents, for which the chosen words unfortunately are now felt It was determined that these terms mayto have been somewhat misleading. It has been renamed herein as Engineering Evaluation and Assessment to more closely describe for clarification of the work that has actually been performed.

Finally, the following is a list of key assumptions or premises used in the development of MRP-227.

1. The 1995 Statements of Consideration related to the revised License Renewal Rule (60 FR 22488) address the relationship of license renewal to plant licensing bases. In amending the "first principle of license renewal", the SOC states:

   "*The first principle of license renewal was that, with the exception of age-related degradation unique to license renewal and possibly a few other issues related to safety only during the period of extended operation of nuclear power plants, the regulatory process is adequate to ensure that the licensing bases of all currently operating plants provides and maintains an acceptable level of safety so that operation will not be inimical to public health and safety or common defense and security.*"

   The 1995 SOC also states:

   "*An applicant for license renewal should rely on the plant's CLB, actual plant-specific experience, industry-wide operating experience, as appropriate, and existing engineering evaluations to determine those nonsafety-related systems, structures, and components that are the initial focus of the license renewal review. Consideration of hypothetical failures that could result from system interdependencies that are not part of the CLB and that have not been previously experienced is not required.*

   Therefore, when considering aging management, only the CLB need be considered. Hypothetical failures associated with system interdependencies are not required to be considered in demonstrating adequate aging management. Therefore, the escalation effects were not directly considered in the FMECA process, nor were they required to be considered.

2. Inservice inspection and testing requirements of the ASME Boiler and Pressure Vessel Code (Section XI) and other operating experience (OE) related requirements, when combined with existing regulations, have been adequate to demonstrate continued safe operation and component integrity through 40 years of operation with existing programs.

3. Components not subject to significant aging-related degradation will continue to be managed by the existing programs that are in place (e.g. Section XI and other OE-related requirements), as appropriate. Simply stated, when MRP-227 concludes "No Additional Measures" are needed, it means that no new actions are needed for that component for the renewal period.

4. The Aging Management Review (AMR) topical reports prepared for B&W, CE and Westinghouse plants during the license renewal process were a basis for the work performed for MRP-227 (BAW-2248A, WCAP-14577-R1-A and CE NPSD-1216).

5. The supporting documents for the Issue Management Tables (MRP-205) were another basis for this work. These tables identified all safety significant issues for all PWR primary loop and internals components.

6. The level of analysis and evaluation detail is consistent with the guidance for Systems Structures and Components (SSC) covered in the license renewal Standard Review Plan (NUREG-1800) and in the GALL (NUREG-1801).

7. Consistent with the License Renewal Rule, the current design bases are considered adequate. In the extended operating period, for passive long-lived components, components are screened to determine if they are subject to degradation associated with aging.

8. Components were designed, manufactured, installed and inspected to accepted regulatory standards. In light of the positive operating experience, there is additional validation that the manufacturing and construction processes were adequate.

9. MRP-227 is a living document, which will be periodically updated to reflect both positive and potentially negative information from inspection results obtained by a series of plants entering the period of extended operation.

1.0 Step 1. Identify PWR internals components, materials, and environments

The first step of the process was to identify the PWR internals components and items within the scope of the program on a generic basis. The starting point for the listing of reactor internals components was the IMTs published in MRP-156 and MRP-157 and other existing reports that provided information beneficial to screening. This initial list was augmented to provide additional clarification for plant-to-plant variations in design and materials.

1.1 B&W

AREVA began with a review of BAW-2248A for the seven B&W-design operating units. BAW-2248A is a B&WOG topical report that contains a technical evaluation of aging effects related to B&W PWR internals component items. It was provided to the NRC staff to demonstrate that the effects of aging during the period of extended operation for B&W PWR internals can be adequately managed. The evaluation applies to the following units:

- Arkansas Nuclear One, Unit 1 (ANO-1)
- Oconee Nuclear Station, Units 1, 2, and 3 (ONS-1, -2, -3)
- Three Mile Island, Unit 1 (TMI-1)

The staff provided a review of the topical report (BAW-2248) against the requirements in 10CFR54 and issued a Safety Evaluation Report (SER) in 1999, which resulted in issuance of BAW-2248A in March 2000. Since that time, the B&WOG has disbanded and EPRI, through the MRP, has continued the investigation on potential aging effects and establishment of monitoring and inspection programs for PWR internals component items. (Note: This was contained in BAW-2248A as applicant action item 4.) This The MRP work expanded the effort on a generic basis for all seven operating B&W-design units. Therefore, the MRP work includes not only the five units above, but it now includes the following additional units:

- Crystal River, Unit 3 (CR-3)
- Davis-Besse, Unit 1 (DB-1)

As part of the MRP effort to identify the PWR internals components and items for all of the B&W design units, MRP-157 was used as the starting point and a review of original B&W design drawings was also performed. The MRP-157 report (Table 4-14) contains the listing of B&W PWR internals components and items, which was developed from the original B&WOG report (BAW-2248A) and augmented through personal knowledge and additional record searching for the remaining units not included in the B&WOG report. This effort encompasses each of the components and items in BAW-2248A and MRP-157, and identified a few more items than contained in BAW-2248A and MRP-157. In addition, the MRP effort reviewed and evaluated weld locations associated with all identified internals components. These Therefore,are included in MRP-189, particularly the weld locations (MRP-189 Rev. 1 contains the complete listing of components and items that was used in this step to be used in development of the MRP-227 I&E guidelines).

1.2 CE & W

The complete list of 120 Westinghouse reactor internals components considered in the development of the MRP-227 recommendations is provided in MRP-191 Table 4-4. The NRC has previously accepted the list of 24 structures and components provided in WCAP-14577-R1-A as an acceptable basis for the scope of an aging management review of Westinghouse reactor internals. The list of components developed under the MRP efforts encompasses the same scope as the previous aging management review, but includesadds additional detail and specificity to aid in the aging assessment.

The CE reactor internal component list was also based on the IMT presented in MRP-156. The complete list of 79 CE internals components considered in the development of the MRP-227 recommendations is provided in MRP-191 Table 4-5.

2.0 Step 2. Identify degradation screening criteria

The second step of the process was to develop and apply screening criteria to identify those PWR internals component items for which the effects of age-related degradation on functionality during the license renewal term may be significant. The screening criteria definition agreed upon by the industry expert panel for the MRP is as follows:

- Screening Value – the level of susceptibility when an aging effect may be significant with respect to continued functionality or safety

The screening value was chosen to be sufficiently conservative such that potential component items could be selected for further evaluation of the effects of aging degradation on functionality.

Eight degradation mechanisms are currently considered relevant when assessing material aging in reactor internals (see Section 1.4 of MRP-175). Those degradation mechanisms are:

Stress Corrosion Cracking (SCC),
Irradiation Assisted Stress Corrosion Cracking (IASCC),
Wear,
Fatigue,
Thermal Embrittlement,
Irradiation Embrittlement,
Void Swelling, and
Irradiation Induced Stress Relaxation/Creep.

Development and justification of the screening criteria required knowledge of the specific aging mechanisms and their effects, some engineering judgment, extensive test data, and the use of empirical extrapolation where test data were lacking. The screening criteria used to identify components potentially susceptible to these eight mechanisms and the basis for the screening values is described in detail in MRP-175.

3.0 Step 3. Characterize components and screen for degradation (A, non-A)

The third step in the process is to evaluate the components identified in Step 1 against the screening criteria developed in Step 2 and documented in MRP-175.

3.1 B&W

Tables 3-2 and 3-3 in Section 3 of MRP-189 Rev. 1 contain the results of the initial screening efforts. It should be noted that thermal stress relaxation of austenitic stainless steel bolting was removed as an aging degradation mechanism for the screening process in MRP-189 Rev. 1 as a result of industry discussions and the justification provided in Appendix B of MRP-191. Wear and fatigue that may be related to thermal stress relaxation were likewise removed from consideration for such bolting.

Because of the lack of specific ASME design rules for core support structures at the time of design and construction, Section III of the ASME Code was used as a guideline for the design criteria for the PWR internals in operating B&W units. As noted in BAW-2248A (see cChapter 2 of the report), the qualification of the internals was accomplished by both analytical and test methods. Thus, values of calculated stress, fatigue usage factors, etc. for many of the PWR internals components and items are not available nor were they required at the time of design. Through the expert panel approach, estimates of potential stress, fatigue usage, etc. were made and used for many of the component items during the screening process. Specific stress inputs were only used for screening a limited number of components (MRP-189 Rev. 1 Table 3-2) from existing stress calculations at the time of screening. The loading sources considered in the stress values are discussed in Response to RAI 4-1. For a few items, a review of available records (stress calculation reports, unit-specific analyses, etc.) was performed that was able to identify the various values provided in MRP-189 Rev. 1 Table 3-2 (see Sections 3.2 and 3.3 of MRP-189 Rev. 1).

Table 1 provides the screening parameters for the representative components[2] from each category that are selected for this roadmap discussion, along with the screening results for each of the aging mechanisms and the initial screening category assigned to each component.

Of the B&W RV internals components that were screened-in as "Non-A" in Step 3, 47 components were placed in the "No additional measures" category by Steps 4, 5, 6, 7, and 8. The B&W RV internals was not designed to the ASME Section III, Subsection NG, and no core support structure or internals structure designations were specified by B&W during the design. However, the safety significance of the RV internals components was evaluated for the MRP-157 report and for MRP-190. The safety significance of these 47 components is summarized below.

FMECA Safety Consequence:

Of the 47 components,

- Two have a FMECA safety consequence metric of "2".
- 44 have a FMECA safety consequence of metric of "1"
- Safety consequence for one component (the upper grid assembly rib section) was not evaluated by FMECA as the CUF value used for screening-in fatigue was from the 205-FA design and was considered incorrect for the B&W 177-FA design by the FMECA panel. [Note: This component has an IMT safety consequence of "G" in MRP-157. See below.]

MRP-190 (FMECA) safety consequences metrics:

1. Safe: no or minor hazard condition exists
2. Marginal: safe shutdown is possible (though with reduced margins to adequately cool the core and/or successfully insert the control rods); localized fuel assembly damage
3. Severe: safe shutdown is possible (though with very reduced margins to adequately cool the core and/or successfully insert the control rods); core damage (multiple damaged fuel assemblies)
4. Critical: safe shutdown is not possible (margins to adequately cool the core and/or successfully insert control rods are totally eroded); extensive core damage

IMT Safety Consequence

Of the 47 components,

- Five have IMT safety consequence metrics of "G and F"
- 23 have an IMT safety consequence metric of "G"
- 19 have no IMT safety consequence

MRP-157 (IMT) consequences of failure metrics:

[2] Note: Each of the steps contains information and/or tables that refer to specific tables or sections in the reference documents for the B&W design. A complete listing of components for the B&W design can be found in these tables or sections in the reference documents from which these representative components have been selected for the discussions in this roadmap.

(A) Precludes the ability to reach safe shutdown
(B) Causes a design basis accident
(C) Causes significant onsite and/or offsite exposure
(D) Jeopardizes personnel safety
(E) Breaches reactor coolant pressure boundary
(F) Breaches fuel cladding
(G) Causes a significant economic impact

Therefore, in summary, of the 47 components placed in the "No additional measures" category, none are considered to have any safety related consequence in the event of loss of function from any age-related degradation mechanism.

Table 1
Screening Parameters, Screening Results for Each Aging Mechanism and Initial Screening Category forSelected B&W RI Components (extracted from Tables 3-2 and 3-3 of MRP-189 Rev. 1)

Component	Temp (°F)	Fluence, (n/cm2, >1 MeV)	60-year dpa	Operating Stress (ksi)	Cold Work ≥ 20%	Multi-Pass Weld	CUF	SCC	IASCC	Irradiation SR	Wear	Fatigue	Thermal Embrittle	Irradiation Embrittle	Void Swelling	Initial Screen Category
CRGT Spacer Castings	605	< 5E18	<0.01	10.58	No	No	Assume <0.1	A	A	A	A	A	Not A	A	A	Not A
CRGT Control Rod Guide Tubes	605	< 5E18	<0.01	Assume <30	No	No	Assume <0.1	A	A	A	Not A	A	A	A	A	Not A
CRGT Control Rod Guide Sectors	605	< 5E18	<0.01	Assume <30	No	No	Assume <0.1	A	A	A	Not A	A	A	A	A	Not A
CSS Vent Valve Top and Bottom Retaining Rings	605	< 5E18	<0.01	9.8	No	No	Assume <0.1	A	A	A	A	A	Not A	A	A	Not A
CSS Vent Valve Disc	605	< 5E18	<0.01	Assume <30	No	No	Assume <0.1	A	A	A	A	A	Not A	A	A	Not A
CSS Vent Valve Disc Shaft or Hinge Pin	605	< 5E18	<0.01	Assume <30	No	No	Assume <0.1	A	A	A	A	A	Not A	A	A	Not A
Core Barrel Cylinder	620	5.0E+21	7.5	1.0	No	Yes	0.21	Not A	A	A	A	Not A	A	Not A	A	Not A
Baffle Plates	646	6.4E+22	96	<20	No	No	<0.1	A	Not A	A	A	A	A	Not A	Not A	Not A
Former Plates	647	5.0E+22	75	<20	No	No	<0.1	A	Not A	A	A	A	A	Not A	Not A	Not A
Core Barrel-to-Former Plate Dowels	633	1.5E+22	22.5	Assume <30	No	No	Assume <0.1	A	A	A	A	A	A	Not A	Not A	Not A

Component	Temp (°F)	Fluence, (n/cm2, >1 MeV)	60-year dpa	Operating Stress (ksi)	Cold Work ≥ 20%	Multi-Pass Weld	CUF	SCC	IASCC	Irradiation SR	Wear	Fatigue	Thermal Embrittle	Irradiation Embrittle	Void Swelling	Initial Screen Category
Lower Grid Support Post Cap Screw	560	2.8E+21	4.2	Assume <30	No	No	Assume <0.1	A	A	Not A	Not A	Not A	A	Not A	A	Not A
Flow Distributor (FD) Bolts	560	5.0E+18	0.008	82	No	No	Assume <0.1	Not A	A	A	A	A	A	A	A	Not A

3.2 CE & WW&CE

Design representative values of the key screening parameters for each reactor internals component in the CE and W fleet were required to complete the screening evaluation. A detailed analysis to generate specific values for either the CE or W design was not performed as part of the MRP project. Representative values, meant to be limiting values for the fleet were determined from existing design basis analysis wherever possible. When hard numbers were not available, teams of reactor internals engineering experts were assembled to provide conservative estimates or to determine if there was any potential for the component to exceed the screening criteria. In all cases, the component condition was conservatively estimated. The process used by Westinghouse to determine these values is described in the following subsections. From this information, the team assessed the data for each component and reached consensus on representative values to use in the screening. This process was published in Section 4 of MRP-191. The component conditions as determined by the teams of experts are provided in MRP-191 Table A-1.

The screening process simply compared the estimated component conditions to the MRP-175 screening levels. Based on this screening process, 48 of the 120 Westinghouse components and 8 of the 79 CE components were identified with no potential aging considering each of the degradation mechanisms. The components with no screened-in aging degradation mechanisms are identified in MRP-191 Table 6-5 and Table 6-6 for W and CE components respectively. These components, which are listed in Table 2 and Table 3 of this roadmap document were tentatively placed in Category A, pending review by the FMECA panel in the following step of the assessment process.

Table 2 Westinghouse Components with No Screened-In Degradation Mechanisms (Data extracted from MRP-191 Table 6-5)

Assembly	Sub-Assembly	Component	Material	IMT Conseq. of Failure
Upper Internals Assembly	Control Rod Guide Tube Assemblies and Flow Downcomers	Anti-rotation studs and nuts	304 SS	G
		Bolts	316 SS	NONE
		Flexureless inserts	304 SS	G
		Housing plates	304 SS	G
		Inserts	304 SS	N/A
		Lock bars	304 SS	NONE
		Support pin cover plates	304 SS	NONE
		Support pin cover plate cap screws	316 SS	NONE
		Support pin cover plate locking caps and tie straps	304 SS	NONE
		Support pin nuts	X-750	NONE
		Support pin nuts	316 SS	NONE
		Water flow slot ligaments	304 SS	N/A
	Upper Instrumentation Conduit and Supports	Bolting	316 SS	NONE
		Brackets, clamps, terminal blocks, and conduit straps	304 SS	NONE
		Conduit seal assembly–body, tubesheets	304 SS	NONE

Assembly	Sub-Assembly	Component	Material	IMT Conseq. of Failure
		Conduit seal assembly–tubes	304 SS	NONE
		Conduits	304 SS	NONE
		Flange bases	304 SS	NONE
		Locking caps	304 SS	NONE
		Support tubes	304 SS	NONE
	Upper Plenum	UHI flow columns	304 SS	G
	Upper Support Column Assemblies	Adapters	304 SS	G
		Column bodies	304 SS	G
		Flanges	304 SS	G
		Lock keys	304 SS	G
		Nuts	304 SS	G
	Upper Support Plate Assembly	Bolts	316 SS	NONE
	Upper Support Plate Assembly	Flange	304 SS	N/A
		Lock keys	316 SS	NONE
		Ribs	304 SS	G
		Upper support plate	304 SS	G
Lower Internals Assembly	Bottom Mounted Instrumentation (BMI) Column Assemblies	BMI column lock caps	304L SS	NONE
	Diffuser Plate	Diffuser plate	304 SS	NONE
	Head Cooling Spray Nozzles	Head cooling spray nozzles	304 SS	NONE
	Lower Support Column Assemblies	Lower support column nuts	304 SS	G
		Lower support column sleeves	304 SS	G

Assembly	Sub-Assembly	Component	Material	IMT Conseq. of Failure
	Lower Support Casting or Forging	Lower support forging	304 SS	A, G
	Radial Support Keys	Radial support key lock keys	304 SS	G
	Secondary Core Support (SCS) Assembly	SCS bolts	316 SS	NONE
		SCS energy absorber	304 SS	NONE
		SCS guide post	304 SS	NONE
		SCS housing	304 SS	NONE
		SCS lock keys	304 SS	NONE
Interfacing Components	Interfacing Components	Clevis insert lock keys	Alloy 600	G
		Clevis insert lock keys	316 SS	G
		Head and vessel alignment pin bolts	316 SS	NONE
		Head and vessel alignment pin lock cups	304L SS	NONE
		Head and vessel alignment pins	304 SS	NONE

IMT Consequence of Failure - G: Causes significant economic impact
A: Precludes a safe shutdown

Table 3 CE Components with No Screened-In Degradation Mechanisms (Data extracted from MRP-191 Table 6-6)

Assembly/ Sub-Assembly	Component	Material	IMT Conseq. Of Failure
Upper Internals Assembly	Control rod shroud-bolts	316 SS	N/A
	GSSS studs	316 SS	N/A
	GSSS spherical washer sets	UNS S21800	N/A
	Flange block shear pins	A286 SS	N/A
Control Element Assembly (CEA)–Shroud Assemblies	Shim bolts	316 SS	N/A
Core Support Barrel Assembly	Core barrel snubber lug bolts	316 SS	N/A
	Core barrel snubber lug bolts	A286 SS	N/A
	Alignment key dowel pins	304 SS	NONE

4.0 Step 4. Failure Modes, Effects and Criticality Analysis (FMECA)

The fourth step in the process was to perform a Failure Modes, Effects and Criticality Analysis (FMECA). While the specific approach used by AREVA for the B&W units varied with that used by Westinghouse for the CE and W units, the principles employed were similar and produced conservative results. It is important to note that items that were screened as "A" in step 3 above (i.e. – no augmented aging management needed) were re-assessed and this confirmed that the original screening was valid. A summary of each approach is described below. The details of the approaches are described in MRP-190 for the B&W units and MRP-191 for the CE and W units.

4.1 B&W

The objective of the FMECA, described in detail in MRP-190, is to provide a systematic, qualitative review of the B&W-designed PWR internals to identify combinations of internals component items and age-related degradation mechanisms that potentially result in degradation leading to significant risk. The FMECA is used to examine the susceptibility, and safety and economic consequences of identified internals component item/age-related degradation mechanism combinations. For those items screened as "A" (in Step 3 above), the FMECA team provided verification that there were "no credible degradation mechanisms" associated with these items.

The FMECA approach uses inductive reasoning to ensure that the potential failure of each component item is analyzed to determine the results or effects thereof on the system and to classify each potential failure mode according to its severity.

Each failure mode (i.e., aging effect) was judged on its importance to risk, based on the susceptibility (likelihood of the degradation mechanism) and severity of consequences. For this FMECA, consequences were examined from two perspectives: safety and economic. The FMECA report developed a risk matrix to correlate the consequence severity of a particular age-related degradation mechanism with the susceptibility of that particular mechanism occurring. Different risk bands were used within the matrix to categorize the level of risk of a particular component item/degradation mechanism pair, and provide guidance on the strategies that should be developed to reduce the corresponding risk and a basis for ranking and categorization. This "risk metric" is not to be confused with risk in a probabilistic risk assessment, for which the metrics of core damage frequency and large early release frequency are typically used.

The criticality metrics of a particular component item failure are evaluated qualitatively by assessing both the susceptibility to an age-related degradation mechanism and subsequent effect, and the severity of the consequences (see Figure 4-1 of MRP-189 Rev. 1). For this FMECA, two types of consequences are considered: safety and economic. When considered together, the criticality metrics represent the risk due to the failure of a particular component item. The criticality metrics are fully described in both MRP-189 Rev. 1 and MRP-190 (also see Step 5 below).

4.2 W and CE & W

A FMECA was conducted to evaluate the likelihood and severity of damage associated with the identified degradation mechanism. The Westinghouse FMECA team was asked to review and concur with information for all 120 identified reactor internals components. Similarly the CE FMECA team was asked to review and concur with information for all 79 identified components. While the screening process evaluated only the potential susceptibility of the component to the eight identified aging degradation mechanisms, the FMECA panel considered both the susceptibility and the potential safety consequences of degradation.

The Westinghouse FMECA process and results are described in MRP-191 and summarized in the following sub-sections. The discussion record of the FMECA expert panel meetings is considered Westinghouse proprietary, but can be made available for NRC review.

4.2.1 FMECA Review of Components with No Identified Degradation Mechanism

The evaluation team was charged to review the results for the 48 Westinghouse and 8 CE components with no identified degradation mechanisms. The panel was asked to concur with these screening results or to recommend reinstating the component for further evaluation. The panel concluded that the application of the screening process was extremely conservative and there was no need to reinstate additional components for further evaluation.

The FMECA panel was also asked to review the 48 Westinghouse and 8 CE components with no identified degradation mechanism and determine that there was "No need to assess damage probability". As part of this process, the FMECA panel reviewed the consequences of failure conclusions from the MRP Issue Management Table (IMT) as described in MRP-156. These

IMT consequences are noted in Table 2 and Table 3. The IMT treats consideration of the probability of degradation and the consequences of failure as completely independent phenomena.

4.2.2 Westinghouse NSSS

Of the 48 Westinghouse components considered, the only component with potential safety-related consequence of failure identified in the IMT was the lower core support forging. (The cast stainless steel version of this component was screened-in due to thermal embrittlement concerns.) Loss of support due to catastrophic failure of this structure could preclude safe shut down of the reactor. THowever, the FMECA panel could not identify any potential cause or mode of catastrophic failure that would require aging management of this large forging. The inspection required for non-age related degradation of this component is specified in ASME Section XI. Therefore the lower support forging was not reinstated for additional evaluation.

There were no potential safety-related concerns ("Precludes safe shutdown" or "Breaches fuel cladding") identified in the IMT for the remaining 47 Westinghouse components. Potential economic consequences of failure were noted in 17 of the remaining components. The FMECA panel concurred with this conclusion and concluded that there was no need to include these components in the aging management strategy because there are no safety implications to failure and the economic consequences of unanticipated failure are not severe enough to justify the expenditure of resources to manage such low probabilities of occurrence.

4.2.3 CE

It is difficult to produce a one-to-one correspondence between the CE reactor internals component list in MRP-156 and the list in MRP-227 because additional detail has been added to facilitate the evaluations in MRP-227. However a thorough review showed there are no potential safety related concerns identified for the CE reactor internals components listed in Table 3.

4.2.4 FMECA Review of W and CE Components with One or More Identified Degradation Mechanisms

The FMECA process was employed to assess the likelihood of failure and the likelihood of damage in the remaining 72 Westinghouse and 71 CE components. The FMECA process is described in detail in Section 6 of MRP-191. Additionally it is noted that the members of the FMECA were consistent for all discussions for a given NSSS design.

The FMECA process was conducted on a component-by-component basis and the FMECA categorization was based on the cumulative effects of all eight degradation mechanisms in each component. Potential susceptibility to multiple degradation modes was one of the factors considered by the FMECA panel.

The FMECA panel findings for the Westinghouse reactor internals are provided in Table 6-5 and CE reactor internals in Table 6-6 of MRP-191. The FMECA panel discussions included evaluation of design and analysis data and are therefore considered to be Westinghouse

proprietary. The FMECA panel findings are also included on the lists of potentially susceptible components in each degradation mechanism series. It should be noted that the FMECA ranking is conservatively based on the cumulative effect of all degradation modes and may not be an indicator of a specific single degradation mode.

5.0 Step 5. Severity Categorization (A, B, C)

The fifth step of the process was to use the results of the FMECA to categorize each of the component items into the categories A, B, and C. As was the case with the FMECA, the severity categorization processes used by AREVA and Westinghouse varied in their specific steps but accomplished the intended goal. All of the reactor internals were placed into one of three categories based on the significance and severity of the potential degradation. A summary of each approach is described below. The details of the approaches and results are described in MRP-189 Rev. 1 and MRP-190 for the B&W units and MRP-191 for the CE and W units.

The FMECA panels for both AREVA and Westinghouse agreed that the "A" (or Category A) events are deemed so improbable (very, very low likelihood of occurrence) that even if a Level B, C, or D event were to occur, the risk impact would not be significant.

5.1 B&W

Categorization of PWR internals was subsequently performed, based on the screening criteria and the likelihood and severity of safety 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). This is detailed in MRP-189 Rev. 1 and MRP-190.

The criticality metrics used in the AREVA FMECA are as follows:

5.1.1 Susceptibility

The susceptibility metric is a qualitative assessment of the likelihood (expressed as a probability or frequency) that an age-related degradation mechanism might occur, given the existing environmental conditions (e.g., temperature, pressure, fluence, etc.), material properties (type of metal, stress-strain), etc. occurring over the life of a nuclear power unit (up to 60 calendar years, considering license renewal). The susceptibility is unrelated to the consequences, e.g., the component item failure or loss of function. The susceptibility qualitative metric was determined as a result of the expert panel meeting. This criticality metric uses an A, B, C, D scale (increasing frequency).

> A – Improbable: not likely to occur (Category A from the initial screening performed in Chapter 3 is synonymous with this susceptibility metric; the Category A results were reviewed by the FMECA expert panel)
>
> B – Unexpected: not very likely to occur, though possible; conditions are such that the age-related degradation mechanism is not expected to occur very often

C – Infrequent: likely to occur, conditions are such that the age-related degradation mechanism is expected to occur occasionally

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

B/I – The susceptibility is sometimes modified with an "I" to indicate an improbable occurrence over the 60-year time period being considered. For example: B/I indicates an unexpected, but possible, degradation mechanism whose initiation results in a certain state that is not credible (or improbable), e.g., SCC crack leading to a 360 degree weld crack. To carefully distinguish between the different types of likelihood, it is possible (B) to have SCC cracking around a weld, but improbable (I) that such as crack would grow around the weld to the critical crack size needed to fail the weld.

Component item/degradation mechanism pairs identified as improbable are not explicitly evaluated for consequences. However, there are a number of combinations that while identified as improbable will either result in severe consequences, affect the ability to cope with a LOCA, or will require the successful "operation" of the guide lugs. Accordingly, while not classified into a specific risk band, these items, as noted in the footnotes of Table 4-1 (MRP-189 Rev. 1) should never be removed from the current ASME inspection requirements (VT-3).

5.1.2 Severity of Consequences

Severity classifications are assigned to provide a qualitative measure of the potential consequence resulting from a component item failure. For those component item/age-related degradation mechanism pairs for which the susceptibility metric was assigned an "A," i.e., "Category A," there was no subsequent evaluation of the consequence due to the very low (i.e., improbable) event frequency. For the PWR internals FMECA, two aspects of consequences are considered: safety and economic. Thus, there are two columns in the FMECA for which qualitative metrics are assigned. The two sets of severity of consequence qualitative metrics were determined as a result of the expert panel meeting. These criticality metrics use a 1, 2, 3, 4 scale (increasing severity).

For severity of consequences (safety), the qualitative metric has been defined as:

1. Safe: no or minor hazard condition exists
2. Marginal: safe shutdown is possible (though with reduced margins to adequately cool the core and/or successfully insert the control rods); localized fuel assembly damage
3. Severe: safe shutdown is possible (though with very reduced margins to adequately cool the core and/or successfully insert the control rods); core damage (multiple damaged fuel assemblies)
4. Critical: safe shutdown is not possible (margins to adequately cool the core and/or successfully insert control rods are totally eroded); extensive core damage

The safety consequence metric assigned will be the highest value, i.e., bounding consequence, for normal operation or design basis event (transient, LOCA, seismic) when the failure mode is not detectable. Typically, the safety consequences were estimated to be the same for normal operation and a design basis event (when the failure mode is not detectable). Note that there were no severity of consequences (safety) identified with a metric of 4.

For severity of consequences (economic), the qualitative metric has been defined as:

1. No or trivial cost
2. Cost that can be generally handled within the existing unit budget and resources (order of millions of dollars)
3. Cost that exceeds the normal unit budget and resources (order of tens of million dollars)
4. Cost that potentially affects the utility's overall financial health (order of hundreds of million dollars)

Note that the economic consequences assume that the failure mode is discovered through some means, e.g., unit inspection, notification of discovery at another unit site, etc. This is also conservative when assessing the risk. Note that the severity of consequences (economic) metric was not used in assignment of the preliminary Category A, B, and C items.

Based upon the FMECA results, the PWR internals that were potentially the most affected were placed into Category C, while the components that are potentially 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.

The risk matrix in MRP-189 Rev. 1 (Figure 4-1) does not include a column for the susceptibility metric value of "A" because, as noted in MRP-190 (Section 3.2), the "A" (or Category A) events are deemed so improbable (very, very low likelihood of occurrence) that the safety severity of consequence metric was not evaluated, implying that even if there was an adverse consequence, the risk impact would be insignificant. However, to clarify how component items were categorized, the Figure 1 below provides a correlation to the risk matrix (Figure 4-1 of MRP-189 Rev. 1) and also includes a column for Category A items:

Increasing Susceptibility from A to D

Increasing Safety Consequences from 1 to 4

	A	B	C	D
1	A	A	B	B
2	A	B	C	C
3	A	C	C	C
4*	A	C	C	C

Figure 1: Consequence vs. Susceptibility for Ranking *Note: There are no component items in the B&W-design internal with an assigned safety consequence metric equal to 4; therefore, the last row of this figure is not applicable to the MRP effort.

The initial Category A, B, and C results for selected B&W components are provided in Table 4.

Table 4
Initial Category A, B and C Results for Selected B&W Components (Extracted from Tables 4-1 and 4-2, MRP-189 Rev. 1)

Component	Safety Band	Economic Band	A, B, C (MRP189) Rev. 1
CRGT Spacer Castings	I	III	B
CRGT Control Rod Guide Tubes	II	III	B
CRGT Control Rod Guide Sectors	II	III	B
CSS Vent Valve Top and Bottom Retaining Rings	I	III	B
CSS Vent Valve Disc	I	III	B
CSS Vent Valve Disc Shaft or Hinge Pin	I	III	B
Core Barrel Cylinder	I	II	B
	I	III	
Baffle Plates	III	III	C
	II	III	
	II	II	
Former Plates	III	III	C
	II	III	
	III	III	
Core Barrel-to-Former Plate Dowels	II	II	B
	I	I	
Lower Grid Support Post Cap Screw	I	I	B
	I	I	
	I	I	
Flow Distributor (FD) Bolts	II	III	C
	IV	V	

Component	Degradation Mechanism	Safety Band	Economic Band	A, B, C (MRP189 Rev. 1)
CRGT Spacer Castings	TE	I	III	B
CRGT Control Rod Guide Tubes	Wear	II	III	B
CRGT Control Rod Guide Sectors	Wear	II	III	B
CSS Vent Valve Top and Bottom Retaining Rings	TE	I	III	B
CSS Vent Valve Disc	TE	I	III	B
CSS Vent Valve Disc Shaft or Hinge Pin	TE	I	III	B
Core Barrel Cylinder	SCC	I	II	B
	IE	I	III	
Baffle Plates	IASCC	III	III	C
	IE	II	III	
	VS	II	II	
Former Plates	IASCC	III	III	C
	IE	II	III	
	VS	III	III	
Core Barrel-to-Former Plate Dowels	IE	II	II	B
	VS	I	I	
Lower Grid Support Post Cap Screw	Fatigue	I	I	B
	IE	I	I	
	Wear	I	I	
Flow Distributor (FD) Bolts	SCC	IV	V	C

It is also interesting to compare the IMT (MRP-157) results to the FMECA results. For each component item that constitutes part of the PWR internals, consequences of failure evaluations were performed in the IMT considering each of the applicable degradation mechanisms (without regard for existing mitigation strategies). This includes following the logical path from component failure to safe shutdown. The consequences evaluation is considered to be reality-based not design-based, so these evaluations are not related to the design bases of the B&W units. Scenarios that rely on a sequence of low probability events reach to get a failure may be documented as such and the failure evaluation terminated. Systems that must operate correctly to satisfy the defined failure sequence are identified. It is also noted that the evaluations do not consider electrical system failures due to component item degradation (e.g., RCS instrumentation). The expert panel participants are listed in the IMT and represent a broad scope

of expertise in the design and operation of the B&W units. In the IMT, the general approach used in the consequences of failure evaluations was as follows:

- For each component item, consequences of failure evaluations were performed considering all of the applicable degradation mechanisms identified by the MDM. The evaluations assume that the unit is initially at full power steady-state conditions. Assuming failure while the unit is at other Level A service conditions impacts the availability of various systems, the unit conditions, and therefore the sequence of events to safe shutdown.

- Level A conditions other than full power, as well as Level B, C, and D conditions are considered coincident with component degradation that does not require unit shutdown during normal operations. These coincident conditions are not rigorously treated, but are discussed from the perspective of their potential contribution to adverse consequences.

  [For clarification, this means that service level events (Levels B, C, and D) were not superimposed along with gross failure from aging degradation of the component or item under consideration. This is a similar approach to that used in Chapter 15 of the FSAR.]

- The evaluations consider the functions that the component item supports and the impact that the degradation might have on the ability of the reactor vessel internals to continue performing thoseat functions. For instance, through-wall cracking, significant wear (at a location of contact or close tolerance), or embrittlement, could compromise the structural integrity of a component item, so each is considered in the evaluations. If different degradation mechanisms lead to different results, then each is treated individually. Multiple degradation sites are not considered because common mode and/or cascading failures are not in the scope of the project. Loose parts were generically evaluated as well.

The following consequences of failure were evaluated:

A. Precludes the ability to reach safe shutdown
B. Causes a design basis accident
C. Causes significant onsite and/or offsite exposure
D. Jeopardizes personnel safety
E. Breaches reactor coolant pressure boundary
F. Breaches fuel cladding
G. Causes a significant economic impact

As shown in Table 4-14 of the IMT (MRP-157), none of the safety-related consequences of failure (items A-E) were determined to be applicable (similar to the FMECA results) and only consequences of failure items F and G were determined to be applicable to the B&W PWR internals. However, it should be noted that there were differences between the consequence evaluations performed in the IMT and the FMECA. An explanation of the differences is provided in Appendix B of MRP-190.

5.2 CE & W

All of the reactor internals were placed into one of three categories based on the significance and severity of the potential degradation. These three categories were:

Category A: Component items for which aging degradation significance is minimal orand aging effects are below the screening criteria.
Category B: Component items above screening levels but are not "lead" component items and aging degradation significance is moderate.
Category C: "Lead" component items for which aging degradation significance is high or moderate and aging effects are above screening levels.

5.2.1 Components Placed in Category A Based on FMECA

After review and confirmation by the FMECA panel, all of the components that were not identified in the screening process for potential susceptibility to any of the eight degradation mechanisms were retained as originally placed in Category A.

The FMECA panel also observed that, due to the conservative nature of the screening process, many components that had been identified for potential degradation were known to not be susceptible to degradation. The most obvious example of the conservative nature of the process was that the surveillance capsule components were identified for irradiation embrittlement because the screening process attributed the peak core barrel fluence to all of the potential attachments. However the FMECA panel observed that the surveillance capsules contain dosimetry packages and the fluences were known to be well below the threshold for irradiation embrittlement.

To more accurately reflect the degradation potential for the components and account for the overly conservative nature of the screening process, the FMECA panel recommended that components with low failure likelihood and either low or medium damage likelihood, especially where the potential for any damage was considered to be readily detectable and manageable in attaining a safe operational state, be moved to Category A. Components with low failure likelihood and high damage likelihood were not considered as candidates to be moved to Category A under any conditions. These criteria are illustrated in Figure 2. By definition, all components with potential safety concerns were classified as high damage likelihood. Therefore, no components with identified safety concerns were affected by this re-classification.

Failure Likelihood	Consequence (Damage Likelihood)		
	Low	Medium	High
High	2	3	3
Medium	1	2	3
Low	1	1	2
None	0	0	0

Category A

Figure 2 FMECA Criteria for Aging Significance Table

The 41 Westinghouse components with one or more identified degradation mechanisms that were moved to Category A based on the FMECA results are listed in Table 5. The 48 CE components moved to Category A based on the FMECA are listed in Table 6. The FMECA panel identified 27 Westinghouse and 27 CE components with low failure probability and low damage consequence. There were an additional 14 Westinghouse and 21 CE components with low failure probability and medium damage consequence. Although the FMECA panel identified a potential economic consequence of failure in the components with medium likelihood of damage, the low failure probability resulted in minimal risk to plant operation. Therefore these 14 Westinghouse and 21 CE components were also placed in Category A. Application of the FMECA process to the Lower Core Plate Fuel Alignment Pin Bolts is provided in Example 1.

Example 1: Lower Core Plate Fuel Alignment Pin Bolts Placed In Category A Based on FMECA

Original screening results: MRP-191 Table 5-1

- **IASCC, Wear, Fatigue, Irradiation Embrittlement, Void Swelling, Irradiation Induced Stress Relaxation/Creep**

Functional Description: MRP-191 Section C.2.1

- **The LCP is bolted at the periphery to a ring welded to the ID of the core barrel. The span of the plate is supported by lower support columns that are attached at their lower end to the lower support plate. At the center, a removable plate is provided for access to the vessel lower head region.**

FMECA Conclusion: MRP-191 Table 6-5

- **Low Failure Probability, Low Consequence**
  - **Screening process overestimated fluence because it assumed components attached to LCP saw same peak fluence. These bolts are located on periphery.**
  - **No history of failures**
  - **Bolts are redundant fasteners.**

Table 5. Westinghouse Components Moved to Category A Based on FMECA Process (Data extracted from MRP-191 Table 6-5)

Assembly	Sub-Assembly	Component	Material	IMT Conseq. of Failure	Screened-in Degradation Mechanisms	Likelihood of Failure L, M, H	Likelihood of Damage L, M, H
Upper Internals Assembly	Control Rod Guide Tube Assemblies and Flow Downcomers	Enclosure pins	304 SS	NONE	SCC, Wear	L	M
		Upper guide tube enclosures	304 SS	NONE	SCC, Wear	L	M
		Flanges-intermediate	304 SS	G	SCC, Fatigue	L	M
		Flanges-intermediate	CF8	G	SCC, Fatigue, TE	L	M
		Flanges-lower	304 SS	G	SCC, Fatigue	L	M
		Guide tube support pins	316 SS	NONE	Wear, Fatigue, ISR	L	M
	Mixing Devices	Mixing devices	CF8	NONE	SCC, TE, ISR	L	L
	Upper Core Plate and Fuel Alignment Pins	Fuel alignment pins	316 SS	NONE	Wear	L	L
		Upper core plate	304 SS	A, G	Wear, Fatigue	L	M
	Upper Plenum	UHI flow column bases	CF8	G	TE, IE	L	L
	Upper Support Column Assemblies	Bolts	316 SS	G	Wear, Fatigue, ISR	L	M
		Column bases	CF8	G	SCC, TE, IE	L	M
		Extension tubes	304 SS	G	SCC	L	M
	Upper Support Plate Assembly	Deep beam ribs	304 SS	G	SCC	L	M
		Deep beam stiffeners	304 SS	G	SCC	L	M

Assembly	Sub-Assembly	Component	Material	IMT Conseq. of Failure	Screened-in Degradation Mechanisms	Likelihood of Failure	Likelihood of Damage
						L, M, H	L, M, H
		Inverted top hat (ITH) flange	304 SS	N/A	SCC, Fatigue	L	M
		Inverted top hat (ITH) upper support plate	304 SS	N/A	SCC	L	M
Lower Internals Assembly	Baffle and Former Assembly	Baffle bolting lock bars	304 SS	NONE	IASCC, IE, VS	L	L
	Bottom Mounted Instrumentation (BMI) Column Assemblies	BMI column bolts	316 SS	NONE	Fatigue	L	L
		BMI column extension bars	304 SS	G	IASCC, IE, VS	L	L
		BMI column nuts	304 SS	NONE	IASCC,Wear, Fatigue, IE, VS, ISR	L	L
	Irradiation Specimen Guides	Irradiation specimen guides	304 SS	NONE	Wear, IE	L	L
		Irradiation specimen guide bolts	316 SS	NONE	IASCC, Wear, Fatigue, IE, ISR	L	L
		Irradiation specimen guide lock caps	304L SS	NONE	IE	L	L
		Specimen plugs	304 SS	NONE	IE	L	L
	Lower Core Plate and Fuel Alignment Pins	Fuel alignment pins	316 SS	NONE	IASCC, Wear,IE, VS	L	L
		LCP-fuel alignment pin bolts	316 SS	NONE	IASCC, Wear, Fatigue, IE, VS, ISR	L	L
		LCP-fuel alignment pin lock caps	304L SS	NONE	IASCC, IE, VS	L	L

Assembly	Sub-Assembly	Component	Material	IMT Conseq. of Failure	Screened-in Degradation Mechanisms	Likelihood of Failure	Likelihood of Damage
						L, M, H	L, M, H
	Neutron Panels/Thermal Shield	Neutron panel bolts	316 SS	NONE	IASCC, Wear, Fatigue, IE, ISR	L	L
		Neutron panel lock caps	304 SS	NONE	IE	L	L
		Thermal shield bolts	316 SS	NONE	IASCC, Wear, Fatigue, IE, ISR	L	L
		Thermal shield dowels	316 SS	NONE	IE	L	L
		Thermal shield or neutron panels	304 SS	G	IE	L	L
	Radial Support Keys	Radial support key bolts	304 SS	G	Wear	L	L
	Radial Support Keys	Radial support keys	304 SS	G	SCC, Wear	L	L
	Secondary Core Support (SCS) Assembly	SCS base plate	304 SS	NONE	SCC	L	L
Interfacing Components	Interfacing Components	Clevis inserts	Alloy 600	G	Wear	L	L
		Clevis inserts	304 SS	G	Wear	L	L
		Clevis inserts	Stellite	G	Wear	L	L
		Internals hold-down spring	304 SS	G	Wear	L	L
		Internals hold-down spring	403 SS	G	Wear, TE	L	L

IMT Consequence of Failure - G: Causes significant economic impact
A: Precludes a safe shutdown

Table 6. CE Components Moved to Category A Based on FMECA Process (Data extracted from MRP-191 Table 6-6)

Assembly/ Sub-Assembly	Component	Material	IMT Conseq. of Failure	Screened-in Degradation Mechanisms	Likelihood of Failure	Likelihood of Damage
					L,M,H	L,M,H
Upper Internals Assembly	Upper guide structure support plate	304 SS	G	SCC	L	M
	Upper guide structure support flange-upper	304 SS	G	SCC, Wear	L	M
	Upper guide structure support flange-lower	304 SS	G	SCC	L	M
	Cylindrical skirt	304 SS	G	SCC	L	M
	Grid plate	304 SS	G	SCC	L	M
	Control rod shroud-grid ring	304 SS	N/A	SCC	L	M
	Control rod shroud-grid beams	304 SS	N/A	SCC	L	M
	Control rod shroud-cross braces	304 SS	N/A	SCC	L	M
	GSSS guide structure plate	304 SS	N/A	SCC	L	M
	GSSS support cylinder	304 SS	N/A	SCC	L	M
	Flange blocks	304 SS	N/A	Wear	L	L

Assembly/ Sub-Assembly	Component	Material	IMT Conseq. of Failure	Screened-in Degradation Mechanisms	Likelihood of Failure	Likelihood of Damage
					L,M,H	L,M,H
	Flange block bolts	410 SS	N/A	TE	L	L
	RVLMS support structure tubes	304 SS	N/A	SCC, Wear, Fatigue	L	L
	Fuel bundle guide pins	316 SS	N/A	Wear, Fatigue, ISR	L	L
	Fuel bundle guide pin nuts	304 SS	N/A	Wear, Fatigue, ISR	L	L
	Hold down ring	403 SS/ F6NM	G	Wear, TE	L	L
	Belleville washer	Alloy 718	N/A	Wear	L	L
Lower Support Structure	Core support plate bolts	316 SS	N/A	IASCC, Wear, Fatigue, IE, ISR	L	L
	Core support plate dowel pins	304 SS	N/A	IE	L	L
	Anchor block bolts	316 SS	N/A	Wear, Fatigue, IE, ISR	L	L
	Anchor block dowel pins	304 SS	N/A	IE	L	L
	Fuel alignment pins	304 SS	NONE	IE	L	M
	Core support beams	304 SS	A, G	SCC, Wear	L	L
	Bottom plate	304 SS	N/A	SCC	L	L
	ICI support columns	304 SS	N/A	SCC	L	L

Assembly/ Sub-Assembly	Component	Material	IMT Conseq. of Failure	Screened-in Degradation Mechanisms	Likelihood of Failure	Likelihood of Damage
					L,M,H	L,M,H
Control Element Assembly (CEA)–Shroud Assemblies	CEA shrouds	304 SS	G	SCC	L	M
	CEA shrouds	CPF8/CF8	G	SCC, TE	L	M
	CEA shroud bases	304 SS	G	SCC	L	M
	CEA shroud bases	CF8	G	SCC, TE	L	M
	CEA shroud extension shaft guides	304 SS	G	SCC	L	M
	Modified CEA shroud extension shaft guides	CF8	G	SCC, TE	L	M
	Internal/external spanner nuts	304 SS	NONE	SCC	L	M
	CEA shroud bolts	A286 SS	NONE	Wear, Fatigue, ISR	L	M
	CEA shroud tie rods	304 SS	N/A	SCC	L	M
	Snubber blocks	304 SS	N/A	SCC	L	L
	Snubber shims	XM-29	N/A	Wear	L	L
Core Support Barrel Assembly	Core barrel snubber lugs	304, 321 or 348 SS	G	SCC, Wear	L	L
	Alignment keys	A286 SS	NONE	Wear	L	L
	Alignment keys	304 SS	NONE	Wear	L	L
	Core barrel outlet nozzles	304 SS	G	SCC, Wear	L	M

Assembly/ Sub-Assembly	Component	Material	IMT Conseq. of Failure	Screened-in Degradation Mechanisms	Likelihood of Failure	Likelihood of Damage
					L,M,H	L,M,H
	Thermal shield	304 SS	G	SCC	L	L
	Thermal shield support pins	304 SS	NONE	Wear	L	L
Core Shroud Assembly	Guide lugs	304 or 348 SS	NONE	SCC	L	L
	Guide lug inserts	304, 321 or 348 SS	NONE	Wear	L	L
In-Core Instrumentation (ICI)	ICI guide tubes	316 SS	NONE	SCC, IE	L	L
	ICI nozzle support plate	304 SS	G	SCC	L	L
	ICI thimble support plate	304 SS	G	SCC, Wear	L	L
	ICI thimble tubes-upper	304 SS	NONE	SCC, Wear	L	L

5.2.2 Components Placed in Categories B and C

The remaining 31 Westinghouse and 23 CE "non-Category A" components were evaluated and placed in Category B or Category C based on the FMECA results and analysis using the Category definitions. Each component was assigned a FMECA aging significance grouping based on the FMECA categories as indicated in Figure 2.

Two exceptions were noted to the components identified by the screening and FMECA process. First, it was observed that the X-750 flexures in Westinghouse plants were obsolete due to plant modifications to resolve the aging concerns. These flexures were removed from subsequent consideration. Second, it was noted that the Zr-4 thimble tubes in the CE In-Core Instrumentation system were known to be subject to an irradiation growth phenomenon that was not addressed as one of the eight degradation modes. These thimble tubes were automatically placed in Category C.

Of the remaining components, 12 Westinghouse and 13 CE components ranked as medium failure likelihood and low failure consequence were automatically placed in Category B. Evaluations of the impact of each of the identified degradation mechanisms were used to rank the significance of the remaining 19 Westinghouse and 9 CE components. Based on that ranking, 12 Westinghouse components were identified as Category C and an additional 6 Westinghouse components were added to the Category B list. A total of 6 CE components (including the Zr-4 thimble tubes mentioned above) were identified as Category C, with the remaining 4 components added to Category B.

There were two additional exceptions to this categorization process discussed in Section 7.2 of MRP-191:

1. *The Westinghouse lower support casting, had been identified as a FMECA Group 2 component based on the consequences of an assumed failure. However, consistent with the MRP-134 definitions, this component was placed into Category A after consideration of the very low probability of degradation and consequence due to the identified thermal embrittlement degradation mechanism.*

2. *The otherOne exception is the internals hold down spring fabricated from 304 SS. Thermal "ratcheting", leading to permanent deformation, is not one of the explicitly characterized degradation mechanisms from MRP-175 but may occur in this component and reduce the spring hold-down force over time. This particular phenomenon was assessed to have a moderate likelihood of occurrence; hence, it was assigned to Category B to warrant attention during the development of Inspection and Evaluation (I&E) guidelines.*

The final list of 31 Westinghouse and 23 CE Category B and Category C items is provided in MRP-191 Tables 7-2 and 7-3. This information is summarized here in Tables 6 and 7. This list of Category B and C Components is carried forward into MRP-227 Tables 3-2 and 3-3. The aging management strategy for the reactor internals is built around examination of these items.