CE Guide Lug Insert Bolts. Table 3-2 lists wear, fatigue and stress relaxation as "Existing Programs" for the CE Guide Lug Insert Bolts. Based on the following excerpt from MRP-232, stress relaxation and fatigue could have been moved to "No Additional Measures". However based on the observation that wear is identified as the expected manifestation for all three degradation mechanisms and the ASME Section XI exam already inspects this location for wear, it was determined that the wear inspection provided appropriate aging management for these three mechanisms.

Text from MRP-232 Section 4.1.5:

In several plant designs, the guide lug insert bolts are additionally retained by the tight clearance between the fuel alignment plate keyway and the contact face of the guide lug insert. If all of the guide lug insert bolts were to become loose, the most visible sign of degradation would be "exacerbated wear" in the vicinity of the fuel alignment plate keyway. Abnormal wear in this region would not be a clear indication of guide lug insert bolt stress relaxation, and a specific evaluation of the wear and guide lug insert integrity would be required to identify if stress relaxation or insert bolt cracking were an issue. Guide lug inserts and/or the mating face in the fuel alignment plate keyway were hard-faced during fabrication to resist wear degradation in these components, so loss of structural alignment as a result of such wear is not plausible.

The likelihood of stress relaxation is very low, such that wear and fatigue cracking of these bolts is not likely, so this component is considered to require No Additional Measures. However, since current VT-3 inspection during the 10-year ISI is capable of evaluating the condition of wear on this surface, this component also falls under the classification of Existing. Noise monitoring should also continue to be used as a supplementary method to determine if excessive vibration within the vicinity of the fuel alignment plate is occurring during operation.

CE Fuel Alignment Pins. Table 3-2 lists IASCC, Wear, Fatigue, irradiation embrittlement and stress relaxation as "Existing Programs" for the CE Fuel Alignment Pins. These concerns are directly relevant to the designs with full height shroud panels. The evaluation in MRP-232 recommended an inspection for cracking as the appropriate means of detecting IASCC, fatigue and stress relaxation (cracking of tabs). Aging management for irradiation embrittlement is typically incorporated as an evaluation requirement for the cracking mechanisms.

Wear is a potential concern in all fuel alignment pin designs. It has been listed separately as an existing program for the non-full height shroud panel plants.

Text from MRP-232 Section 4.1.4:

Fuel alignment pins are considered as components with an Existing inspection, because the fuel alignment pins are inspected as part of the normal 10-year in-service inspection.

The concerns with IASCC, wear, and irradiation-induced stress relaxation from the screening analysis have been dispositioned and loss of functionality has been determined to be a very low probability event. The adverse effects of stress relaxation should be eliminated by the welded tabs, as long as the tabs remain intact. A visual inspection of the tabs may be required to assure their integrity and to justify reliance on these tabs to disposition the effects of stress-relaxation. Visual inspection of the welded tabs would be recommended as a one-time inspection to be completed during the 10-year in-service inspection (ISI) prior to entering the period of plant life extension. Missing tabs, missing fuel alignment pins, or abnormal wear of the fuel alignment pins would be easy to evaluate using a VT-3 visual inspection.

Westinghouse Upper Core Plate Alignment Pins. Table 3-3 lists SCC and wear as "Existing Programs" for the Westinghouse Upper Core Plate Alignment Pins. The first visible manifestation of the degradation effect for these two mechanisms was determined to be wear. The aging management recommendation for both SCC and wear is to inspect for wear. A failure due to SCC might also be reported as a result of a VT-3 exam. The aging management program is based on inspecting for wear as the early indicator of an aging concern.

Text from MRP-232 Section 4.2.9.2:

Upper core plate alignment pins are fabricated from austenitic stainless steel and are welded into place. Based on the structural welds that hold these pins in position and the potential for localized residual stresses from welding to exceed the screening threshold for stress, they were conservatively screened in for SCC. However cracking of these components is considered unlikely and the first visible manifestation of damage is expected to be wear.

Westinghouse Core Barrel Flange. Table 3-3 lists SCC as an "Expansion Program" and wear as an "Existing Program" for the Westinghouse Core Barrel Flange. Wear of the core barrel flange is listed in the Existing Programs Table for Westinghouse. However the designation of the core barrel flange as an expansion requirement for SCC may appear circular as the Primary item is the Core Barrel Flange Weld. The intent was to expand to the remaining core barrel welds. This is clear in the MRP-227 section 4 Tables. Table 3-3 is potentially confusing in this regard.

Text from MRP-232 Section 4.2.2.7:

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.

The proposed inspection methods are appropriate for degradation when cracking is the primary effect. The cracking-related mechanisms would include SCC, IASCC and fatigue. The VT-3 examination can also be used to detect visible signs of wear. Gross deformation due to swelling may also be detectable in a visual exam, but severe effects of swelling may occur long before the deformation is observable. However, there is no non-destructive inspection technique capable of detecting thermal or irradiation embrittlement. At this time there is no practical way to monitor stress relaxation by measuring loads in reactor internal bolting. Although MRP-227 has identified irradiation embrittlement, thermal embrittlement, void swelling and irradiation induced stress relaxation as primary or expansion degradation mechanisms for multiple components in Tables 3-2 and 3-4, there are no effective inspections techniques for these mechanisms. Although there are no inspection requirements for these components the proposed Rev.0 would include these mechanisms listed under the effect "Aging Management" in Table 4-2, 4-3, 4-5, 4-6, 4-8 and 4-9.

The aging management strategies for void swelling and stress relaxation must rely on detection of the secondary consequences of these mechanisms. The irradiation aging analysis conducted on the baffle-former structure provides the basis for determining these consequences. The aging analysis does suggest relative displacement along seams in the baffle structure that may be directly observable. The only other observable consequence of void swelling in the baffle-former-barrel assembly is IASCC failure of baffle-former bolts and baffle-edge bolts caused by swelling in the former plates. The timing of the failure is affected by compensating loss of load due to stress relaxation. Therefore, inspections of the bolting systems for IASCC failure provide an indicator of these related degradation mechanisms.

The aging management strategies for thermal embrittlement and irradiation embrittlement rely largely on trend curves compiled from laboratory data. Embrittlement can lead to loss of toughness that reduces the flaw tolerance of the materials. This loss of toughness can have a drastic effect on the acceptable flaw size in the component. Section 6.2.2 of MRP-227 provides guidance on fracture mechanics analysis of irradiated components. Because the irradiated components and thermally embrittled components have a reduced flaw tolerance, it is particularly important that any active cracking mechanism in these components be actively managed. In the inspection strategy outlined in Tables 38-49, every component with an identified embrittlement concern has a corresponding requirement for inspection related to one or more potential cracking mechanism.

The second question in this RAI concerns component/aging mechanism combinations that are in Tables 3-1 through 3-3 but not identified in Tables 4-1 through 4-9.

For the B&W design, Table 4-1 lists "overload" for the core barrel assembly locking devices and locking welds of the baffle-to-former bolts and internal baffle-to-baffle bolts; this was inadvertently not captured in Table 3-1. MRP-227 Table 3-1 will be revised to include reference to "Note 1" for these components and the note will be revised to include the locking devices for these components. This is the only example of a component/aging mechanism combination in Table 3-1 not identified in Tables 4-1 or 4-4 for the B&W design components.

The CE and Westinghouse Tables, Tables 4-2, 4-3, 4-5, 4-6, 4-8 and 4-9, will be updated in the '-A' version of MRP-227 to include the appropriate component/aging mechanisms combinations (see Appendix A to these RAI responses).

RAI 4-18: As a follow-up to RAI-26 (second set of RAIs), please clarify if components that are predicted to locally exceed 5% swelling by volume are inspected for cracking at those locations. Provide justification why any such components that exceed this criterion are not recommended for inspection.

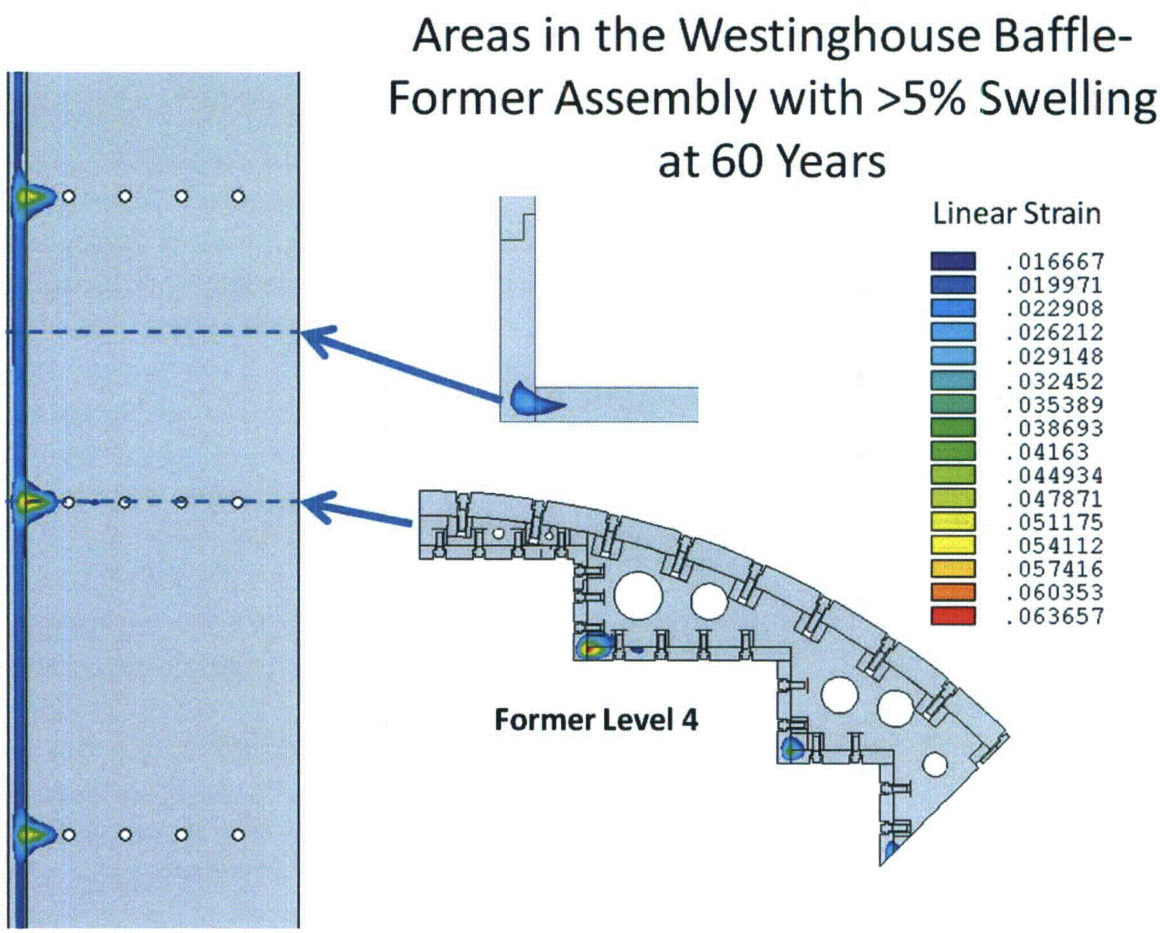
Response: For the B&W design, there are no locations predicted to locally exceed 5% swelling by volume. Fro the CE and Westinghouse designs, the MRP-227 I&E Guidelines monitor the distortion caused by the integrated effects of void swelling over the entire volume of a PWR internals assembly. The inspections that implement this requirement are shown in Tables 4-2 and 4-3 of MRP-227 for Combustion Engineering and Westinghouse plants, respectively. The inspections focus on the regions of high swelling and locations where the accumulated effects of swelling over the assembly are most likely to be visible. The particular distortions that led to the recommendations were the result of the functionality analyses reported in MRP-229 and MRP-230.

These functionality analyses showed that the volumes of material for which relatively high "effective irradiation strain growth" was calculated were very small and localized, to regions of combined high fluence, high gamma heating, and limited cooling. (Note: the volumetric swelling is equal to 3x the irradiation growth strain.) In the Westinghouse design, the high swelling regions were limited to high fluence seams where there are joints between the baffle and former plates. Typical Westinghouse high fluence regions are indicated in the accompanying figure. MRP-227 requires VT-3 examinations of the baffle-former assembly and the baffle-edge bolts at these locations. Even though the FEA aging evaluation included these levels of swelling, the analysis did not indicate stresses high enough to initiate IASCC in these components. The high swelling locations are remote from significant structural load paths and cracking at these locations is not expected to have a significant structural impact. Therefore it was determined that the

VT-3 examinations were adequate to manage cracking due to age related degradation at these locations.

A similar situation occurs near the mating surfaces of the upper and lower core shroud former plates in the CE design (MRP-227 Figure 4-14). The MRP-227 guidelines require inspections of the joint for observable distortion and the adjacent welds for cracking. Because the welds are considered a structural element, the requirement is for an EVT-1 inspection at the joint.

The MRP continues to sponsor projects to investigate the expected significance of void swelling. If detrimental effects due to localized void swelling are observed in operating reactors, the MRP and industry will then develop specific guidance to characterize its effects, recommend appropriate changes to inspections, and determine its future acceptability.



Note: 0.01667 Linear Strain is equivalent to 5% Volumetric Swelling

RAI 4-19: The following components, listed as an example only, were originally identified for potential aging degradation but they were dispositioned under "No Measures" category. The staff requests that MRP provide an explanation for not performing any analysis prior to binning them under "No Measures" category.

COMBUSTION ENGINEERING COMPONENTS

Component	Aging Effect	MRP-232 – Reference
Core Support Plate Bolts	Irradiation Embrittlement	Table 2-11
Fuel Alignment Pins (304 stainless steels)	Irradiation Embrittlement	Table 2-11
Core Shroud Tie Rods	Irradiation Embrittlement	Table 2-11
Core Shroud Tie Rods	Irradiation Induced Stress Relaxation	Table 2-16
Core Support Plate	IASCC, Wear	Tables 2-3 and 2-5

WESTINGHOUSE COMPONENTS

Component	Aging Effect	MRP-232 – Reference
Lower Core Plate Fuel Alignment Pin Bolts	Irradiation Embrittlement	Table 2-12
Lower Core Plate Fuel Alignment Pin Bolts	Irradiation Induced Stress Relaxation	Table 2-17
Bottom Mounted Instrumentation (BMI) Column Bodies	IASCC, Irradiation Embrittlement	Tables 2-4 and 2-12
BMI Column Cruciforms	IASCC, Thermal Embrittlement, and Irradiation Embrittlement	Tables 2-4. 2-10 and 2-12

Response: A separate roadmap document has been developed to augment this and several other RAI responses, and is included as part of the overall response package. In addition to the description of the eight-step process that was used to develop MRP-227, this document also points to the details of the MRP-227 supporting documents, including details on components in Tables 3-1, 3-2, and 3-3 that were originally screened in as non-Category A for at least one of the eight age-related degradation mechanisms that were found through subsequent evaluations to require No Additional Measures.

As a specific example of the way in which this roadmap document was prepared, one of the items in the RAI 4-19 table – core shroud tie rods in CE plants subject to irradiation embrittlement and irradiation-induced stress relaxation – will be followed through the entire categorization process. Core shroud tie rods in CE plants were originally screened in (see Table 5-2 in MRP-191) for wear, fatigue, irradiation embrittlement, and irradiation-induced stress relaxation. For this reason, the original categorization based only on screening with respect to potential significance of the eight age-related degradation mechanisms was non-Category A. Based on the FMECA evaluation (see Table 6-6 in MRP-191), the FMECA group to which core shroud tie rods was Group 1, the lowest of the three groups, based on likelihood of failure and consequences of that failure. Then, Table 7-3 in MRP-191 shows that the preliminary placement of Category B, which means that the core shroud tie rods are not considered as a candidate for a Primary component, but could be considered as a candidate for an Expansion component. Then, in Table 4-4 of MRP-232, core shroud tie rods are shown as non-Category A for the same four age-related mechanisms as before (wear, fatigue, irradiation embrittlement, and irradiation-induced stress relaxation), but is also shown as requiring No Additional Measures. The discussion and justification of this recommendation takes place in the last paragraph of Section 4.1.1 of MRP-232, where the results of finite element analysis and associated engineering assessment show that the combined effects of irradiation embrittlement and irradiation-induced stress relaxation do not indicate the need for any additional examinations beyond those required for core support structures, and that the remaining effects of wear and fatigue also do not reach the threshold for further consideration.

Similar types of documentation are available for the other items in the RAI table as well as the B&W design, and that reasoning is provided by the accompanying roadmap.

RAI 4-20: Many licensees have incorporated ANS 51.1, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," which categorizes transient events in a classification scheme by condition, into facility licensing bases. According to the standard, an acceptance criterion for a Condition II event is that by itself, a Condition II incident cannot generate a more serious incident of the Condition III or IV category without other incidents occurring independently or result in a consequential loss of function of the reactor coolant system or reactor containment barriers. For example, an anticipated operational occurrence, such as a turbine trip from full power, should not cause a degraded component inside the reactor vessel to fail in such a way that a control element assembly ejection could occur. Further detailed discussion regarding this criterion is available in NRC Regulatory Issue Summary 2005-29, "Anticipated Transients that Could Develop Into More Serious Events."

For those components that the FMECA or functionality analyses provided a basis to reduce or eliminate inspection requirements, address whether consideration of this "non-escalation" criterion affects this basis.

> **Response:** The non-escalation criterion in ANS 51.1 is intended to assure that a relatively frequent Condition II transient cannot generate a more serious (and less frequent) Condition III or Condition IV event without other incidents occurring independently. At issue is whether or not the FMECA evaluations and the functionality analyses took into account the full range of potential consequences of a relatively frequent operational transient causing failure of an aging-degraded component, including the possibility of triggering a much less frequent but more serious event. The answer is that the FMECA evaluations certainly considered such possibilities, with the intent to determine the most conservative range of consequences of aging-degraded component failure.
>
> To be very precise about this point, the first two steps performed in the MRP-227 development process are described with the issue of consequences emphasized. For example, the first step (screening) evaluated potential component susceptibility to the eight age-related degradation mechanisms and their effects, with no consideration of consequences. The results of this step were documented in MRP-189 and MRP-191. However, in the second step, the FMECA process used expert elicitation to determine the most conservative set of consequences from the failure of a degraded component, regardless of whether that failure was initiated by a Condition II transient or by some other event. The results of this step were documented in MRP-190 and MRP-191.
>
> In this regard, the experts had access to previous work by the industry where the consequences of loose parts caused by component failures were systematically evaluated, including both safety and economic consequences. The expert elicitation process used this information, along with the expert opinions about the consequences of many other postulated failure modes for aged components, as the basis for the assignment of risk consequences. These risk consequences included both the potential for that postulated component failure to escalate into a more serious event, or the observation that the postulated component failure resulted in modest to negligible risk (non-escalation). For a component that was found to be potentially susceptible to one or more of the eight age-related degradation effects, but for which the consequences of postulated failure were found to be negligible, it could be inferred that the non-escalation of consequences was a

contributing factor to reduced or eliminated inspection requirements. For a component that was found to be potentially susceptible to one or more of the eight age-related degradation effects, and for which the consequences of postulated failure were found to be significant, the escalation of consequences could be inferred to be a contributing factor for the assignment of Primary, Expansion, or Existing Program inspection requirements.

In a few cases, the evaluation of potential escalation was formalized explicitly, as illustrated by the following excerpt from Section 3.3 of MRP-190:

"During the meeting, it was recognized that if the failure mode was not detectable, a second consequence question needed to be posed: would the degradation mechanism result in a more severe consequence (if undetected) when a design basis event occurred (e.g., seismic or LOCA)? The consequence column metric is the most conservative consequence (between normal operation and consideration of a design basis event, when needed)."

Following this logic, Section 3.4.2 of MRP-190 describes a number of these "cascading failures," as shown in Tables 5-2 through 5-6 of MRP-190. Similar treatment of cascading consequences was carried out in the preparation of MRP-191; however, the process is not explicitly described.

The functionality analyses results described in MRP-229 and MRP-230 were used, in part, to support the recommendations in MRP-231 and MRP-232 that were eventually incorporated into the requirements of MRP-227. These functionality analyses were primarily carried out to evaluate the degree of aging related degradation in the identified components. The focus of the functionality analysis was to determine the effects of long term exposure to normal operating conditions (including normal operating transients) on component condition. There was no consideration of off-normal transients in the functionality analyses. Even so, the functionality analyses provided evidence for non-susceptibility in a few cases – i.e., clear evidence that 60 years of conservative operation resulted in insignificant degradation. Such findings led to the determination that no additional measures were required to monitor aging effects in these components. Stated another way, the insignificant degradation did not affect the ability of the component to perform its intended function, including reasonable assurance that safety-related functions would be maintained.

Therefore, degradation of the reactor internals leading to a failure that could initiate a system transient at any level is considered in MRP-227 as a potential loss of functionality requiring aging management. Analysis of the consequences of the transient or more serious events that could subsequently develop is beyond the scope of the document

RAI 4-21: Address the effects of failures of uninspected components and components with failure modes that aren't detectable during normal operations (i.e., undetectable failure modes) through the following considerations:

a) Discuss whether the failure of any such component(s) could be an initiating event for a plant transient or other accident.

b) Discuss the effect of failure of any such component(s) on system performance assuming a design basis event (i.e., plant transients, accidents, and seismic events representative of upset, emergency and faulted loading conditions) occurs prior to mitigating the failure. As part of this discussion, describe any analysis that has been performed, or any plant-specific analysis that is needed, to demonstrate that acceptable system design margins are retained under this scenario.

Finally, discuss whether the final recommendation not to inspect these components is affected by addressing the scenarios described in a) and b) above.

Response: The RAI concerns PWR internals components for which no inspection requirements are prescribed in MRP-227. There are two sets of such components. The first and largest set of such components are the components that fall into the category of No Additional Measures and which are not core support structures that continue to require periodic visual examination based upon ASME Code Section XI requirements. The second set is a relatively small set of components that are essentially inaccessible without extraordinary measures, but which are sufficiently susceptible to one or more aging degradation effects that they were placed in the Expansion category. We will refer to the first set as No Additional Measures/Non-Code (NAM/NC) components. We will refer to the second set as Inaccessible Expansion (INEX) components.

With respect to Part a) of the RAI, all components – including NAM/NC and INEX components were evaluated during the FMECA expert elicitation process to determine the consequences of their postulated failure, and the evaluation considered the potential for that failure to be an initiating event for a plant transient or other accident. Discussions of these evaluations are documented in MRP-190 (Section 3 and the tables in Appendix A) and MRP-191 (Section X). Special attention should be paid to the discussion of "cascading events" in MRP-190. In particular, the FMECA exercises reported in MRP-190 and MRP-191 both took advantage of previous industry efforts to evaluate the consequences of "loose parts' caused by PWR internals component failure, regardless of the cause of such failures.

With respect to Part b) of the RAI, the expert elicitation process used in the FMECA evaluations documented in MRP-190 (Sections 3.2, 3.3, and 3.4) and MRP-191 (Section Y) did include the consequences of design-basis events that could occur following postulated failure of a component, including NAM/NC and INEX components. The consequences derived from the expert elicitation process were qualitative but conservative, and were based upon the participating expert's knowledge and experience with the systems being considered. In some cases, generic or plant-specific analyses that attempted to simulate the consequences of the postulated failures were also available for consideration.

With respect to the final part of the RAI, the results of the FMECA evaluations for the NAM/NC components showed that even the most conservative estimates did not cause unacceptable safety or economic consequences. Therefore, even with the inclusion of the scenarios described in Part a) and Part b) of the RAI, the final recommendation in MRP-227 was unaffected. This was not the case for the INEX components. For this set of components, the final recommendation – as shown by example in Table 4-4 on Pages 4-27 and 4-28 (Expansion Components for B&W plants) – inspection was not an option (short of disassembly) because of inaccessibility. Therefore, the final recommendation was to permit either engineering evaluation or component replacement as options. It should be noted that, for the choice of engineering evaluation, plant-specific analysis considering the full range of design-basis loadings would be required.

RAI 4-22: MRP-190, Section 3 discusses component failure modes that aren't detectable during normal operations (i.e., undetectable failure modes). Provide specific examples of important components that are susceptible to these failure modes. Describe any special consideration or weighting that components susceptible to these failure modes received in either the FMECA (e.g., through the failure severity rankings) or the final MRP-227 inspection recommendations (e.g., by elevating the component to the primary inspection category) given that the component failure may not be discovered until the next refueling outage (i.e., up to 2 years after failure occurs). Provide specific examples to illustrate the process used to evaluate these components.

Response: Section 3.3 of MRP-190 describes a step in the FMECA expert elicitation process where the experts were asked to consider the potential for an undetected component failure mode and the modification of the resulting consequences if a design-basis event, such as an earthquake or a LOCA were to occur prior to mitigation:

"During the meeting, it was recognized that if the failure mode was not detectable, a second consequence question needed to be posed: would the degradation mechanism result in a more severe consequence (if undetected) when a design basis event occurred (e.g., seismic or LOCA)? The consequence column metric is the most conservative

consequence (between normal operation and consideration of a design basis event, when needed)."

The results of that modification to the FMECA expert elicitation process are found in Table A-1 of MRP-190, with particular attention to Column 10 (Heading: Detectable) and Column 11 (Heading: Comments). An excellent pair of examples is provided by the Plenum Cover Assembly bottom flange and the Plenum Cover Assembly support flange. Both of these components received a "No" in Column 10. However, in the former case, the "No" includes a comment that ultrasonic examination (UT) during a periodic ten-year inspection would provide an adequate basis for managing the degradation, whereas the latter presented no operational issues. Another excellent example is the Plenum Cover Assembly top flange, which falls into the same category as the Plenum Cover Assembly bottom flange.

The information contained in Table A-1 of MRP-190 carried over to MRP-231, which provided the actual inspection recommendations for inclusion in MRP-227; however, the fundamental evaluation results and discussion are contained in the columns of Table A-1 of MRP-190.

The RAI refers specifically to MRP-190, which applies to B&W designs. MRP-191, which applies to CE and Westinghouse designs, does not describe a similar approach for addressing the potential for a component failure mode that would be undetected during normal operations. However, the FMECA process steps described in MRP-191 contain the essential elements that would have led to results similar to those reported in MRP-190. For example, on page 6-1 of MRP-191, one of the six basic questions to be addressed by the FMECA expert panel was: "How might the failure be detected?" The range of expertise on the panel would have been able to identify those component failure modes that would be undetected during normal plant operation, and would have been able to adjust the worst-case consequences accordingly. These worst-case consequences would then be reflected in the severity rankings which, in turn, would have been reflected in the recommendations for inspection in MRP-232, if warranted.

RAI 4-23: Identify any components that should be replaced either prior to the period of extended operation or during the period of extended operation because they may not be able to perform their intended function during design basis events (normal, transient, emergency and faulted conditions) based on the results of the FMECA or functionality assessment.

Response: No internals components for Babcock & Wilcox, Combustion Engineering, or Westinghouse PWR plants were identified as requiring replacement due to inability to perform their intended function during design basis events prior to the period of extended operation or during the period of extended operation, as the result of either FMECA or functionality assessment.

RAI 4-24: Tables 2-18 and 2-19 in MRP-232 and Table 3-8 in MRP-231 indicate that a licensee's aging management program will inspect CE, Westinghouse and B&W RVI components for thermally or irradiation-enhanced stress relaxation. However, various CE, Westinghouse and B&W RVI components that are susceptible to thermally and irradiation-enhanced stress relaxation have been downgraded from Categories B or C to the "No Additional Measures" Category.

Document the basis of the evaluation that was utilized to downgrade these components to the "No Additional Measures" Category. Demonstrate that both inspected and uninspected components susceptible to thermal or irradiation-enhanced stress relaxation maintain their design function during emergency and faulted events postulated at the end of the period of extended operation. This demonstration should show that the recommended inspection method is adequate for identifying or assessing stress relaxation before design margins become inadequate.

If a generic evaluation of the adequacy of such components under design basis loading is not possible, identify plant-specific action items that must be performed by licensees to ensure these components will be able to maintain their design function during design emergency and faulted conditions at the end of the period of extended operation.

In particular, identify the projected loss of preload due to stress relaxation at the end of the period of extended operation for the following bolts.

- Combustion Engineering---Core Support Column Bolts; Core Shroud Bolts; Guide Lug Insert Bolts; Barrel-Core Shroud Bolts
- Westinghouse----Baffle-edge Bolts; Baffle-former Bolts; Lower Support Column Bolts
- Babcock and Wilcox-----Baffle-to-Baffle Bolts; Core Barrel-to-Former Bolts; Baffle-to-Former Bolts.

Explain why this loss in preload will not result in the loss of the intended function for these bolts during design basis events that are postulated at the end of the period of extended operation.

Response: The technical basis for downgrading to No Additional Measures some PWR internals components that were originally screened in as either Category B (moderately susceptible) or Category C (significantly susceptible) to thermally-induced or irradiation-enhanced stress relaxation will be addressed in a separate "roadmap" response. That response will cover the first three paragraphs of this RAI.

The last part of the RAI requests a quantitative "projected loss of preload due to stress relaxation at the end of the period of extended operation" for a list of specified bolts in Combustion Engineering, Westinghouse, and B&W plants, with the intent to "ensure that these components will be able to maintain their design function during design emergency and faulted conditions at the end of the period of extended operation." No such calculations were carried out on either a generic or design-specific basis during the development of MRP-227 or any of its supporting documents. Therefore, it is not possible to respond to the specific request for additional information.

However, the development of MRP-227 and its supporting documents did address several aspects pertinent to the underlying intent of the RAI, which will be discussed in the following paragraphs.

Thermally-Induced Stress Relaxation. This topic was addressed thoroughly in the text of MRP-175 and in Appendix B of MRP-191, and that discussion will be briefly summarized here. In the absence of a significant radiation environment, the relaxation of preload for PWR internals is of the order of 10% to 20%. This amount of preload loss is of the same order as the variability of preload in bolted assemblies that are subjected to very careful bolt torque patterns, depending upon the number of torque passes and the flexibility of the underlying structure. The loss of preload from thermally-induced stress relaxation occurs over a period of a few thousand hours, after which further exposure with time essentially shows saturation of the effect, with very little additional loss. Both the loss of preload from thermally-induced stress relaxation and the variability of preload from the torque pattern are accounted for by increasing the specified preload slightly, so that the residual preload is sufficient to maintain function.

Irradiation-Enhanced Stress Relaxation. The functionality analyses of a representative core barrel assembly in a B&W plant, a representative core shroud assembly in a Combustion Engineering plant, and a representative baffle-former-barrel assembly in a Westinghouse plant evaluated bolt preload changes caused by the combination of irradiation-enhanced stress relaxation (or creep) and void swelling. The analyses were based upon 30 years of normal operation in a high-leakage core environment, followed by 30 years of normal operation in a low-leakage core. No design-basis emergency conditions or design-basis faulted conditions were included in the analyses. As a result, the calculated reductions (and increases) in bolt preload cannot be considered to be projected losses of preload due to stress relaxation with the intent to ensure the capability to maintain design function. In addition, the estimates of preload change are highly variable, depending on the location of a particular bolt relative to the core (for radiation exposure) and the location of a particular bolt relative to the prying action caused by integrated void swelling effects and associated global deformation.

MRP-229 and MRP-230 provide the detailed discussion of the functionality analyses of the three representative assemblies listed above, while MRP-231 and MRP-232 summarize the essential results. Suffice it to say that, during the first 30-year period of high-leakage core loading, loss of bolt preload due to irradiation-enhanced stress relaxation tends to dominate any potential increase in bolt preload from prying caused by void swelling, with complete loss of preload in some cases and very large losses in preload (say up to 70% loss of preload) not uncommon. After the second 30-year period, when the combination of effects is more mixed, the results are even more variable, since the prying action caused by the integrated effects of void swelling counteract the effects of irradiation-enhanced relaxation for many bolts. A typical example is provided by Figures 3-54 to 3-63 in MRP-230, which show the calculated baffle-former bolt preload for the representative Westinghouse plant at six year intervals, beginning with the 6th year of high-leakage core loading. The reduction in preload is larger for the bolts nearer the active core region (say rows 3 through 6) than in rows 1 and 8. At year 30, when the transition from a high-leakage core to a low-leakage core takes place, the variability is at its extreme, with a few baffle-former bolts showing the dark blue color signifying essentially complete loss of preload, while others near or slightly above the initial preload. Examining the plots for the period of low-leakage core loading all the way out to 60 years of simulated operation shows that roughly half of the bolts has lost all or very nearly all of their preload, while the top row of baffle-former bolts continues to maintain the initial preload.

Similar results were observed for the representative B&W plant core barrel assembly functionality calculations reported in MRP-229 and evaluated in MRP-231 (see Section 3.2.2.3). The portion of the MRP-229 Section 4 paragraph summarizing the results is repeated here for convenience:
"*Irradiation-induced stress relaxation is most significant for the baffle-to-baffle bolts at internal baffle corners, where relaxation of over 90% (in some case complete loss of bolt load) is typical. The baffle-to-former bolts at former elevations 2 to 7 experience large amounts of stress relaxation of up to 90%. Maximum relaxation of the external baffle-to-baffle and the core barrel-to-former bolts is about 50% and 40%, respectively. In general, however, these bolts experience much smaller magnitudes of stress relaxation.*"

Two other topics are relevant to this discussion. First, bolt locking devices are a significant deterrent to complete bolting failure resulting from complete loss of preload, and the supporting documentation for MRP-227 describes the benefits of bolt locking devices in some detail. Second, examining bolts to determine potential loss of preload is an ongoing research activity that has yet to lead to practical and effective application.

For example, in situ measurement of the length of a bolt by ultrasonic testing to determine its current length relative to its initial preloaded length is a potential technique that has yet to be demonstrated as practical and workable. The accuracy required for the measurement and/or the need to compare to a very accurate baseline measurement represent obstacles that have yet to be overcome. Even the possibility of detecting complete loss of preload through some type of impact-echo system has yet to shown to be cost-effective.

In summary, quantitative calculation of the loss of preload through the end of the renewal period term, along with analytical demonstration that the calculated loss of preload does not prevent bolted assemblies from performing their intended function, is not within the scope of activities carried out in the MRP-227 development program. Instead, the aging management program elements for PWR internals components that are subject to potentially significant irradiation-enhanced stress relaxation or creep are based on detection and management of subsequent aging effects that ensue from the loss of preload, which include wear, fatigue, and IASCC.

RAI 4-25: The effect of radiation on material ductility is a TLAA for B&W vessel internals. Section 4.2.6 of the Three Mile Island Nuclear Station Unit 1 License Renewal Application indicates the following:

> The effects of irradiation on the materials properties and deformation limits for the reactor vessel internals was evaluated for the current licensing basis in Topical report BAW-10008, Revision 1, Appendix E. This analysis concluded that at the end of the forty years, the internals will have adequate ductility to absorb local strains at the regions of maximum stress intensity, and that irradiation will not adversely affect deformation limits. This analysis is a TLAA that will be managed by the PWR Vessel Internals program for the period of extended operation.

The staff requests that the MRP explain how this issue has been addressed for B&W vessel Internals program. Are the effects of radiation on material ductility a TLAA for CE and Westinghouse vessel internals? If that is not the case, provide an explanation for not performing a TLAA evaluation in CE and Westinghouse vessel internals. If it is a TLAA explain how this issue is addressed in PWR vessel internals program.

Response: This question was originally identified as a Time Limited Aging Analysis (TLAA) in Section 5.1.5 of BAW-2248A and in the associated staff Final Safety Evaluation Report (FSER). The TLAA was defined as renewal applicant action item number 12 in the FSER, but as noted in Section 5.1.5 of BAW-2248A, this TLAA for

B&W RV internals is to be resolved on a plant-specific basis per 10CFR 54.21 (c)(1)(iii), based on the results and conclusions of the RV internals aging management program (RVIAMP) discussed in Section 4.6 of BAW-2248A.

As described in the attached roadmap (which is a response for RAI 4-1 and various other RAI responses that have been grouped within it), the B&W Owners Group (B&WOG) disbanded and the ongoing RVIAMP efforts were superseded and are currently being completed through the MRP efforts, which ultimately will establish the appropriate monitoring and inspection programs to be performed (i.e., the MRP-227 requirements).

As stated above in RAI 4-25, the TMI-1 LRA indicates that the PWR Vessel Internals program will manage this TLAA and this was concluded by the staff to be adequate.

However, AREVA, through the MRP, has addressed this TLAA on a generic basis for the B&W RV internals with issuance of a non-proprietary report (AREVA 47-9048125-002, provided to the NRC by MRP Letter 2010-064, dated October 26, 2010), which is available for each of the B&W unit licensees. This document is provided as an attachment to the RAI responses. A proprietary version will be submitted to the NRC by AREVA NP Inc. Corporate Regulatory Affairs with an accompanying affidavit in accordance with 10 CFR2.390(b). Both of these documents are being provided to the Staff as information in support of the MRP-227 review, which clearly show that for the irradiation levels at the end of 60-year lifetime for this component, there will be adequate ductility at operating temperature to absorb local strains at the regions of maximum stress intensity, and that irradiation will not adversely affect deformation limits.

No TLAA related to material ductility has been identified for either CE or Westinghouse reactor internals. However, it should be noted that the effects of loss of fracture toughness and reduced material ductility are managed for both CE and Westinghouse reactor internals through aging management program elements defined in the Section 4 tables of MRP-227, and that these effects will be taken into consideration for any engineering evaluations of detected conditions that exceed the examination acceptance criteria contained in Section 5 of MRP-227.

RAI 4-26: RAI-20 (Set #2) asked about how the linkage between primary and expansion components was determined and how the expansion criteria (i.e., the results of the primary inspection that triggers an expansion inspection) was developed. While the response to RAI-20 is clear and the process is generally understood by staff, there is still a lack of explicit justification for many of the linkages and the explicit expansion criteria. That is, there is not a clear basis why the primary component was selected and why the expansion linkage is both appropriate and comprehensive (i.e., no other components should be linked).

The basis for the criteria used to trigger expansion inspections, and the acceptability of this basis, should also be provided for each of the primary and expansion linkages. As an example, expansion criteria for the core barrel and baffle barrel bolts are not triggered unless there is a 5% or higher failure rate in the baffle former bolts. Similarly, a 10% rate of rejection for either the upper core barrel or lower core barrel bolts triggers the expansion items. The basis for these expansion criteria should demonstrate that the failure rate or rate of rejection specified for the baffle former bolts and the upper core barrel or lower core barrel bolts are sufficient to ensure that significant degradation is not occurring in the expansion components such that the design margin requirements for expansion components and associated systems are satisfied.

Response: The third item (Item 3) in the response to RAI-20 (Set #2) provided a considerable amount of detail on the basis for the linkages between Primary and Expansion components, including a reference to the expert panel elicitations from which the recommendations for those linkages were derived -- "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)". This RAI acknowledges that that response was clear and the process generally understood by staff, so the process will not be described further. The RAI is asking for additional technical support for the expansion criteria themselves other than the results of the expert panel elicitation; i.e., what was the basis for the expert panel to debate and to eventually agree on a specific expansion criterion, such as the 5% failure rate in baffle-former bolts or the 10% failure rate for upper core barrel bolts?

An attempt will be made to respond to the RAI by example, rather than by going through lengthy and perhaps speculative discussions for each of the Primary and Expansion links. Two examples will be provided – one of which deals with expansion criteria for a bolted assembly and the other of which deals with a welded components.

First, the expert elicitation process by which the linkages and criteria were developed took place at a point in time where a draft C of MRP-227 was available (out of eventually a draft J), and early drafts of MRP-231 and MRP-232 were also available (the supporting documents with specific inspection recommendations for MRP-227). Second, preliminary results were also available for functionality analyses of such assemblies as the B&W core support shield assembly, the Combustion Engineering core shroud assembly, and the Westinghouse baffle-former assembly. Third, many of the experts involved in the expert elicitation process were familiar with or had been involved with safety evaluations of internals bolting patterns carried out in the 1990s.

With this back ground in place, consider the particular combination of a Primary component – Core Shroud Assembly core shroud bolts in bolted core shroud Combustion Engineering plants – and the linked Expansion component – Core Shroud Assembly barrel-shroud bolts in bolted core shroud Combustion Engineering plants. The degradation effect is cracking from either fatigue or IASCC, and the environment comparison provides the justification for selecting one set of bolts as Primary and the other set of bolts as Expansion.

The experts discussed a number of potential expansion criteria, starting with a recommendation that, for similar bolting configurations, functionality could be maintained by 50% or fewer of the bolts, with certain restrictions on functional bolt distribution. Some experts argued that observing a much lower number of failures in the core shroud bolts, such as 10%, would be appropriate to indicate that expansion to include barrel-shroud bolts. The argument for the 10% figure was supported by the observation that the 100% of accessible bolts examination requirement for the core shroud bolts would also cover a large number of core shroud bolts with much lower fluence. The discussion among the experts then led to options for considering the number of failed core shroud bolts detected for different core shroud plates with different fluence levels, and whether failures for high-exposure core shroud bolts should carry more influence than failures for lower-exposure core shroud bolts. The consensus that was eventually reached was based on: (1) essentially all of the core shroud bolts are accessible, which means that both high-exposure bolts and low-exposure bolts will be part of the sample space; (2) a relatively large number of core shroud bolts failures can be tolerated in the high-exposure region without compromising assembly function; and (3) functionality is optimally assured by limiting the number of failures in the low-exposure region.

This eventual consensus of the experts led to the recommendation that confirmed failure of a number greater than 5% of core shroud bolts on the four lowest shroud plates at the largest distance from the core, which are the low-exposure shroud bolts, would trigger the examination of the Expansion linkage components. This was felt to be a very conservative recommendation, since the low-exposure core shroud bolts still have roughly five times the exposure of the barrel. The conservatism of the recommendation provided adequate assurance that the specified expansion criterion for the core shroud bolts would ensure that significant degradation would not occur in the barrel-shroud bolts prior to the expansion examinations.

The second example is the combination of a Primary component – the upper core barrel flange weld in Westinghouse plants – and the linked Expansion components – the 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.

<table>
<tr><th>Item</th><th>Applicability</th><th>Effect (Mechanism)</th><th>Primary Link (Note 1)</th><th>Examination Method (Note 1)</th><th>Examination Coverage</th></tr>
<tr><td>Core Barrel Assembly
Upper thermal shield bolts (UTS)</td><td>All plants</td><td rowspan="2">Cracking (SCC)</td><td rowspan="2">UCB and LCB bolts</td><td rowspan="2">Volumetric examination (UT)</td><td rowspan="2">100% of accessible bolts

See Figure 4-7</td></tr>
<tr><td>Core Barrel Assembly
Surveillance specimen holder tube (SSHT) studs/nuts (CR-3) or bolts (DB)</td><td>CR-3, DB</td></tr>
</table>

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

<table>
<tr><th>Item</th><th>Applicability</th><th>Effect (Mechanism)</th><th>Primary Link (Note 1)</th><th>Examination Method (Note 1)</th><th>Examination Coverage</th></tr>
<tr><td>Core Barrel Assembly
Upper thermal shield bolts (UTS) and their locking devices</td><td>All plants</td><td rowspan="2">Bolt: Cracking (SCC)

Locking Devices: Loss of material, damaged, distorted, or missing locking devices (Wear or Fatigue damage by failed bolts)</td><td rowspan="2">UCB and LCB bolts and their locking devices</td><td rowspan="2">Bolt: Volumetric examination (UT).

Locking Devices: Visual (VT-3) examination.</td><td rowspan="2">100% of accessible bolts and locking devices.

See Figure 4-7.</td></tr>
<tr><td>Core Barrel Assembly
Surveillance specimen holder tube (SSHT) studs/nuts (CR-3) or bolts (DB) and their locking devices</td><td>CR-3, DB</td></tr>
</table>

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 See Figure 4-8
Flow Distributor Assembly Flow distributor bolts (FD)					

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. See Figure 4-8.
Flow Distributor Assembly Flow distributor bolts (FD) and their locking devices					

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 l:

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: ~~b~~Baffle ~~p~~plates, ~~b~~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.

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[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:

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[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

MRP-227 Roadmap
October 29, 2010

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)

<table>
<tr><th>Component</th><th>Safety Band</th><th>Economic Band</th><th>A, B, C (MRP189) Rev. 1</th></tr>
<tr><td>CRGT Spacer Castings</td><td>I</td><td>III</td><td>B</td></tr>
<tr><td>CRGT Control Rod Guide Tubes</td><td>II</td><td>III</td><td>B</td></tr>
<tr><td>CRGT Control Rod Guide Sectors</td><td>II</td><td>III</td><td>B</td></tr>
<tr><td>CSS Vent Valve Top and Bottom Retaining Rings</td><td>I</td><td>III</td><td>B</td></tr>
<tr><td>CSS Vent Valve Disc</td><td>I</td><td>III</td><td>B</td></tr>
<tr><td>CSS Vent Valve Disc Shaft or Hinge Pin</td><td>I</td><td>III</td><td>B</td></tr>
<tr><td rowspan="2">Core Barrel Cylinder</td><td>I</td><td>II</td><td rowspan="2">B</td></tr>
<tr><td>I</td><td>III</td></tr>
<tr><td rowspan="3">Baffle Plates</td><td>III</td><td>III</td><td rowspan="3">C</td></tr>
<tr><td>II</td><td>III</td></tr>
<tr><td>II</td><td>II</td></tr>
<tr><td rowspan="3">Former Plates</td><td>III</td><td>III</td><td rowspan="3">C</td></tr>
<tr><td>II</td><td>III</td></tr>
<tr><td>III</td><td>III</td></tr>
<tr><td rowspan="2">Core Barrel-to-Former Plate Dowels</td><td>II</td><td>II</td><td rowspan="2">B</td></tr>
<tr><td>I</td><td>I</td></tr>
<tr><td rowspan="3">Lower Grid Support Post Cap Screw</td><td>I</td><td>I</td><td rowspan="3">B</td></tr>
<tr><td>I</td><td>I</td></tr>
<tr><td>I</td><td>I</td></tr>
<tr><td rowspan="2">Flow Distributor (FD) Bolts</td><td>II</td><td>III</td><td rowspan="2">C</td></tr>
<tr><td>IV</td><td>V</td></tr>
</table>

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), <u>none of the safety-related consequences of failure (items A-E) were determined to be applicable (similar to the FMECA results)</u> 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.