August 30, 2010

Mr. Neil Wilmshurst
Vice President & Chief Nuclear Officer
Electric Power Research Institute
1300 West WT Harris Blvd
Charlotte, NC 28262

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION NUMBER 4, RE: ELECTRIC POWER RESEARCH INSTITUTE TOPICAL REPORT 1016596, "MATERIALS RELIABILITY PROGRAM (MRP): PRESSURIZED WATER REACTOR INTERNALS INSPECTION AND EVALUATION GUIDELINES (MRP-227 – REV. 0)" (TAC NO. ME0680)

Dear Mr. Wilmshurst:

By letter dated January 12, 2009, Electric Power Research Institute submitted for U.S. Nuclear Regulatory Commission (NRC) staff review Topical Report 1016596, "Materials Reliability Program (MRP): Pressurized Water Reactor Internals Inspection and Evaluation Guidelines." Upon review of the information provided to date, the NRC staff has determined that additional information is needed to support completion of the review. During discussions with Chuck Welty, Technical Executive, and Anne Demma, Project Manager, we agreed that the NRC staff will receive your response to the enclosed RAI questions by October 29, 2010.

If you have any questions regarding the enclosed RAI questions, please contact me at 301-415-1847.

Sincerely,

/RA/

Sheldon D. Stuchell, Senior Project Manager
Licensing Processes Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project Nos. 669 and 689

Enclosure:
RAI Questions

cc w/encl: See next page

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Electric Power Research Institute

Project No. 689
Project No. 669

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REQUEST FOR ADDITIONAL INFORMATION (RAI)

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT (TR) 1016596, "MATERIALS RELIABILITY PROGRAM (MRP): PRESSURIZED WATER REACTOR INTERNALS INSPECTION AND EVALUATION GUIDELINES"

(MRP-227 – REV. 0)

ELECTRIC POWER RESEARCH INSTITUTE (EPRI)

PROJECT NO. 669

In a letter dated January 12, 2009, EPRI submitted a TR MRP-227, Rev. 0, "Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines," which addresses an aging management program (AMP) for the PWR reactor vessel internal (RVI) components. During the review process, the NRC staff provided three separate Requests for Additional Information letters, which were responded to. The NRC staff continues its review of TR MRP-227, Rev. 0, and supporting information.

The NRC staff has developed the following set of questions based upon its continuing review of topical report MRP-227, "Pressurized Water Reactor Internal Inspection and Evaluation Guidelines," supporting technical reports provided by the Materials Reliability Program (MRP)/Electric Power Research Institute (EPRI) (about which some of the following questions are directed), and MRP/EPRI responses to prior RAIs.

The staff has concluded that if the following questions can be answered adequately and completely, the staff can move forward and complete its review and safety evaluation (SE) for MRP-227. However, it should be recognized that inadequate or incomplete answers to the following questions may result in the imposition of limitations and conditions on the use of MRP-227 in the staff's SE. In particular, the staff will strongly consider the resolution of any remaining issues which can be addressed as plant-specific action items by the imposition of limitations and conditions within the SE.

1. Develop a comprehensive roadmap that describes how components that were categorized as non-A (i.e., Category B and C components) by the initial screening analysis were binned into the final recommended inspection categories (i.e., primary, expansion, existing, and no additional measures). The roadmap should include, for a sufficient number of components to demonstrate the general practice, the results of the initial screening analysis, the failure modes, effects, and criticality analysis (FMECA) susceptibility and consequence results and rationale supporting the results, a brief

ENCLOSURE 1

summary of the functionality assessment results (if applicable) and a description of the use of these results (i.e., impact) in defining the recommended inspection program, a summary of the recommended inspection program (i.e., inspection type, periodicity, and accessibility requirements), and the supporting basis for this program. As an integral part of this demonstration, ensure that justification/rationale exists for classifying initial B and C components with medium or high failure consequences in any bin other than the primary category. The roadmap should cite references and data sources used to develop information/justification supporting the recommended ranking for each component discussed (e.g., consequence analysis).

The roadmap should identify, for each component discussed, the loading sources that provide the normal operating stresses considered for each component. Loading sources may include, for example, pressure, thermal, deadweight, residual stress (e.g., from fabrication/installation, welding), hydrodynamic, and preload stresses. The roadmap should identify which, if any, of these stresses may produce significant cyclic or transitory stresses under normal operating conditions. Indicate the portion of the normal operating stress due to static loading sources and the portion attributed to significant cyclic or transitory load sources that may contribute to fatigue.

2. Develop a comprehensive roadmap for components that were categorized as A by the initial screening analysis and have medium or high failure consequences. This roadmap should include the consequence results and supporting rationale associated with the recommended categorization. Indicate if and why any of these components were moved into the primary or expansion categories based on either the FMECA or the functionality analysis. The roadmap should cite references and data sources used to develop information/justification supporting the recommended ranking for each component (e.g., consequence analysis).

   The roadmap should identify, for each of the medium or high failure consequences components, the loading sources that provide the normal operating stresses considered for each component. Loading sources may include, for example, pressure, thermal, deadweight, residual stress (e.g., from fabrication/installation, welding), hydrodynamic, and preload stresses. The roadmap should identify which, if any, of these stresses may produce significant cyclic or transitory stresses under normal operating conditions. Indicate the portion of the normal operating stress due to static loading sources and the portion attributed to significant cyclic or transitory load sources that may contribute to fatigue.

3. In addition to the information supplied in response to Questions 1 and 2, the FMECA process should be more fully documented to support its use in the development of the

recommended aging management programs. Discuss and describe the following aspects of the FMECA analysis for both the Westinghouse/CE and B&W studies[1]:

a) Relate the list of experts who participated in the FMECA with the list of required technical specialties to perform this analysis.

b) Describe the FMECA process used in each study including the following items:

- key assumptions,
- scope and motivation (e.g., what components addressed, use to confirm initial screening results, use to develop recommendations for component classification),
- approach (e.g., ranking and consensus process used to obtain results, consideration of either single degradation mechanisms or combined mechanisms, ranking definitions, development of classification matrices, use of classification matrices to develop component classification recommendations, evaluation of the effects and consequences of component failure for Westinghouse, CE, and B&W designs), and
- analysis and results (e.g., how individual estimates were amalgamated to determine final estimate, how ranking biases among the various experts were addressed and reconciled, results of both susceptibility and consequence analyses, the impact of the FMECA on the final severity of component failure rankings with a focus on instances where the FMECA was used to change initial severity rankings).

In particular, identify and then discuss differences between the Westinghouse/CE and B&W FMECA studies. This discussion should address, for example, differences in the susceptibility/failure likelihood and severity/damage likelihood definitions, consideration of the effects of either single or multiple degradation mechanisms, and recommended inspection categories based on the tabulated matrix values. Additionally, this discussion should identify any Westinghouse, CE, and B&W components having a similar or equivalent function that received different aging management program classifications based on the FMECA, and, as appropriate, either document why the differences exist or describe how differences in the Westinghouse/CE and B&W FMECA results for similar components were reconciled.

c) One of the FMECA assumptions states "…no consideration was given to manufacturing errors, maintenance errors, installation errors, transport errors, or any other type of random or human errors." Why aren't these failure modes considered for components that have complex manufacturing, maintenance, or installation

[1] Note that some of this information was presented during the meeting on June 8, 2010 between the industry and the NRC.

procedures? What is the justification for not considering these effects for components with medium or severe failure consequences?

d) Section 6.2 of MRP-191 indicates that a "…FMECA does not serve well to identify multiple failures." What is meant by multiple failures (i.e., common cause, indirect, cascading failures, or another type)? How is this deficiency in the FMECA process addressed as part of the aging management strategy?

e) Section 6.2 of MRP-191 also indicates "…operability, reliability, and availability issues were also considered." This statement is unclear. What explicit issues were considered in the FMECA and how were they considered by the experts? For example, was there an explicit process to factor these issues into the FMECA rankings or were these issues used to determine which of several components with similar degradation mechanisms and likelihood and/or failure consequence would be chosen for the primary inspection category?

4. In addition to the information provided in response to Question 1, discuss the impact of the functionality analysis in terms of determining or modifying the final inspection requirements for components. Specifically, provide an overview of how the functionality analysis impacted the recommended component classifications (i.e., primary, expansion, existing programs, and no additional measures) that were initially developed prior to conducting the functionality analysis. For primary and expansion components indicate how the functionality analysis was used to determine the type of inspection and the inspection periodicity. Finally, identify all similar Westinghouse, CE, and B&W components (i.e., those that perform a similar function and have similar failure consequences) that have different final inspection classification and requirements based on the functionality analysis. Provide justification for any differences that exist.

5. In addition to the information provided in response to Questions 1 and 3, discuss in general how susceptibility to multiple degradation mechanisms was considered when developing the final inspection recommendations. The final inspection category recommendations for each component appear to typically be based on the susceptibility and consequences associated with the single most-dominant degradation mechanism. However, many components are subject to multiple degradation mechanisms and it is not clear how the synergistic effect of multiple degradation mechanisms was considered in the final recommendations. The concern is that a component subjected to multiple degradation mechanisms may be more likely to experience a greater level of total degradation than a component that is subject to a single mechanism (even though the component with the single mechanism may be more susceptible to that mechanism). Discuss the acceptability of the recommended inspection method for primary components that are susceptible to multiple degradation mechanisms. Demonstrate that the recommended method is capable of identifying degradation due to all significant contributing mechanisms (and not just the single most-dominant mechanism) before component design margins are exceeded.

6. Several previous RAIs (e.g., RAIs 2-11, 2-18, and 3-8) have questioned whether plant-specific analyses are required to demonstrate that each plant is appropriately represented by MRP-227 such that the proposed aging management programs (AMPs) are applicable. That is, confirmation that the plant's initial design and operating conditions fall within the scope of the MRP-227 evaluation, the plant complies with important assumptions underlying the MRP-227 analysis, and changes in plant design or operating conditions (e.g., resulting from power uprates) have been appropriately considered. Meeting these conditions is necessary to ensure that the plant-specific AMP inspection requirements (i.e., the primary inspection components, inspection type and periodicity) would not be different from the MRP-227 recommendations determined through more generic evaluation.

   The responses to these various RAIs have indicated that a plant-specific analysis to demonstrate the applicability of MRP-227 guidance is not required because plant-specific differences have been considered by: (a) evaluating operating experience throughout the commercial fleet; (b) using a conservative "out-in" core loading pattern in the functionality analysis; and, (c) assessing several known plant-specific conditions in the FMECA. The responses also justify the representativeness of MRP-227 because: (a) base load operational profiles (i.e., fixed power levels) are similar among plants, and (b) no design changes have been enacted by plants other than those identified in generic industry guidance or recommended by the original nuclear steam system supply vendors. However, given the variability in design and operational conditions that currently exists in PWR plants, the staff is not convinced the MRP-227 AMP requirements are necessarily appropriate for each plant. For instance, it is not clear that the "out-in" core loading pattern is conservative given that some degradation mechanisms do not initiate until low-leakage core conditions are imposed in the functionality model.

   Therefore, the staff requests that guidance be developed that will allow individual licensees to assess the applicability of the MRP-227 method and results. This guidance should particularly focus on demonstrating the applicability of (a) the FMECA and functionality assessments, and (b) the recommended inspection category, inspection method and periodicity for each component. Specifically, this guidance should allow a licensee to determine if plant-specific differences in the RVI design or operating conditions (i.e., power uprate level) result in different component inspection categories (i.e., primary, expansion, existing, and no additional measures) than recommended within MRP-227. Alternatively, additional analysis or justification may be provided to demonstrate that the MRP-227 approach and results are generically applicable such that plant-specific differences in the RVI design or operating conditions do not result in different component inspection categories than recommended within MRP-227.

In the absence of adequate guidance, the staff will consider the need to implement limitations and conditions on the use of MRP-227, which would address plant-specific action items necessary to address this issue for each facility.

7. Provide guidance on the process that should be followed by licensees if the plant-specific application of the MRP-227 guidelines identifies that inspection or aging management of a primary component (i.e., as defined in MRP-227) is not necessary. The guidance should address, for example, the plant-specific criteria and process for reclassifying the aging management program for a primary component, disposition of linked expansion components, and identification of (an) alternative plant-specific primary component(s) to be used in lieu of the generic MRP-227 recommendation for that degradation mechanism.

   The response to this question should specifically propose text that would be added to the "-A" version of MRP-227 to address this issue.

8. This question discusses accessibility requirements for primary and expansion components. Define the appropriate inspection coverage to ensure the component being inspected does not lose its intended function and the process to be followed if the inspection does not meet the inspection coverage. Provide additional guidance on the component accessibility requirements for each primary and expansion component (i.e., those in MRP-227 Tables 4-1 through 4-6, 4-8 and 4-9) such that the results of the inspection can be credited as satisfying the requirements of the aging management program. This guidance should include, at a minimum, the following considerations:

   a) For each component, identify the location(s) where degradation is expected.

   b) Define the appropriate inspection coverage at this location to ensure that enough of the surrounding material is inspected such that there is assurance that the degradation will be identified before it challenges component or system integrity (i.e., the intended function of the component is retained).

   c) Describe the procedure that a licensee should follow if inspection accessibility is insufficient to provide the required inspection coverage or if the inspection does not meet other minimum requirements as specified in MRP-227 and MRP-228.

   This procedure must address providing an appropriate justification for continued operation with the reduced examination requirements to the NRC for review and approval. The guidance should address the process for adjusting the inspection area and/or coverage interval for both welded and non-welded components as a function of the component being inspected and/or the degradation mechanism being assessed during plant-specific inspections.

   With respect to inaccessible components, the MRP should:

a) Identify any components that are; (1) totally inaccessible (cannot be inspected) and (2) the management of their aging effects is dependent on the inspection results from another primary, expansion, or an existing component.

b) For the components identified in "a)," identify the primary, expansion, or other existing components that are the surrogate for the inaccessible components and explain why the surrogates are the limiting components for the aging effects that need management.

c) For the totally inaccessible components or for the inaccessible portion of primary or expansion components, what are the requirements that the licensee must follow to ensure that the components do not lose their intended function as a result of flaws in the accessible components.

The response to this question should specifically propose text that would be added to the "-A" version of MRP-227 to address this issue.

9. A number of components are identified as being covered by existing programs. However, there is no summary of existing RVI programs provided in MRP-227 or supporting documentation. Add a summary of existing programs being credited to MRP-227. If an existing program is consistent with a program definition given in the staff's Generic Aging Lessons Learned (GALL) report, it is sufficient to simply identify the related GALL program definition. For existing programs lacking such a convenient reference, a summary should be provided which describes the following program requirements for an acceptable existing program: (a) scope (i.e., components inspected/monitored), (b) the applicable inspection, monitoring, or testing requirements and acceptance criteria, (c) the periodicity of the program, and (d) any other relevant requirements. This summary should identify the degradation mechanism(s) that are intended to be monitored or mitigated by each existing program and provide justification that each program is sufficient to monitor and/or mitigate all the expected degradation mechanisms identified in MRP-227 for the applicable component(s).

The response to this question should specifically propose text that would be added to the "-A" version of MRP-227 to address this issue.

10. MRP-227 guidance is used to develop component-level aging management programs and inspection requirements. Further, the development of these programs and requirements has not considered the effects of transitory design basis events (DBE) on the performance of degraded components or structures. However, as indicated in the response to RAI 2-16, the current industry expectation is that "...when age-related degradation effects are detected during the examinations specified in MRP-227, the suitability of the degraded component for continued service will necessarily take into consideration the full range of design basis event (DBE) effects." Therefore, staff

believes that guidance and requirements should be provided to ensure that licensees perform consistent plant-specific evaluations of the effects of degraded components under both normal and transitory DBEs (i.e., normal, upset, emergency, and faulted loading conditions). These evaluations should provide reasonable assurance that the systems associated with the degraded components will maintain required design margins and that inspection, repair, and replacement requirements are both adequate and timely. The guidance and requirements should, in part:

a) Identify the number or percentage of related primary or expansion components that should be inspected and the allowable number of degraded or non-functioning components for each system to ensure acceptable performance under DBEs. Discuss the appropriateness of developing generic versus plant-specific inspection requirements for each system. Alternatively, the guidance should describe how the plant-specific analysis should be performed to determine the inspection sample and allowable number of degraded components for each system, and

b) Describe the additional inspection requirements that should be triggered if degraded components are found as part of the primary inspection.

The consideration of item b) should provide guidance for increasing the sample size to inspect other similar components within the system that are subject to the observed degradation mechanism. It should either provide guidance for expanding the inspection to components in other systems that are subject to the same degradation mechanism or justify the adequacy of existing expansion criteria in MRP-227. As an example, this guidance should specify the percentage of baffle-to-former bolts that should be inspected and the percentage that may be degraded before system performance under design basis loading conditions is affected. If degraded baffle-to-former bolts are found, this guidance should next specify the additional baffle-to-former bolts that should be inspected and, for instance, the number of expansion baffle-to-baffle or core barrel-to-former bolts that should be inspected.

11. RAI 2-1 asked for justification of the inspection periodicity recommended in MRP-227 for reactor vessel internal components given that there is little operating experience for basing inspection periodicity and that analysis to evaluate the evolution of degradation in these components has a large degree of uncertainty. The response to that question primarily justifies the adequacy of the recommended inspection intervals based on the functionality analysis, which predicts that degradation will gradually worsen over time and will not suddenly progress. However, the inspection periodicity is not based, as is typically the case, on an evaluation of the maximum level of degradation that is acceptable for components to fulfill their intended design requirements, and the predicted time to reach this level of degradation based on the extent of degradation found during the inspection and evaluation of the rate of degradation with continuing operation. Therefore, the staff requests additional justification for recommended inspection intervals. This justification should address why the current MRP-227

approach is appropriate for determining inspection periodicity and that determining periodicity based on a component level evaluation to ensure that the required component design margins are retained between inspections is not required.

Alternatively, provide information about the plans of licensees to perform the initial primary inspections required by MRP-227. Staff understands that some licensees are planning to inspect all required primary components during the first refueling outage after entering into the license renewal period. Staff therefore seeks to determine if this approach is being adopted by other licensees that have or will shortly enter the license renewal period. Clarify if this approach is either recommended or required within MRP-227.

12. RAI 2-21 asked about the need to develop more definitive acceptance criteria for inspections to ensure uniform interpretation and implementation from plant to plant. The industry response indicated that more definitive criteria in MRP-227 is not needed because the inspectors will receive component-specific training and that any observable degradation will require further disposition through the corrective action process. However, staff remains concerned that this approach is not sufficient given the variability associated with inspection conditions and interpretation of inspection results. Therefore, staff requests that more definitive inspection acceptance criteria be developed for the VT-1, EVT-1, UT, and VT-3 inspections for each of the primary and expansion components. These criteria should be a function of the required accuracy and precision of the particular technique and also the application of this technique to each particular component (i.e., accessibility limitations, expected degradation location, expected degradation type).

    In the absence of adequate guidance, the staff will consider the need to implement limitations and conditions on the use of MRP-227, which would address plant-specific action items necessary to address this issue for each facility.

13. Provide a description of how international and the US operating experience is (or is planned to be) documented, tracked, and updated so that it will support continued refinement of MRP-227 guidance and inform plant-specific aging management programs.

    The staff believes that it would be advisable for documentation regarding the US and international operating experience related to degradation in RVI components be compiled in a single document that could be used to support the process of updating MRP-227.

14. Verify that neither MRP-210, "Materials Reliability Program: Fracture Toughness Evaluation of Highly Irradiated PWR Stainless Steel Internal Components," nor Section 6 of MRP-227 will be used to disposition (i.e., determine need to repair, need to replace, or inspection periodicity) degraded components identified during RVI inspections and that

this guidance will instead be provided by WCAP-17096, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," Revision 2, December 2009. If this is not the case, provide a description of the relationship between MRP-210, Section 6 of MRP-227, and WCAP-17096 and identify the aspects of the disposition analysis that will be governed by each document.

15. There have been several previous RAIs related to cast austenitic stainless steel (CASS) materials, but several questions/issues remain.

    a. The industry is currently supporting using a minimum irradiation embrittlement (IE) threshold of 1 displacement per atom (dpa) to determine susceptibility of CASS components to IE, yet available data seems to support a threshold of 0.3 dpa or less. There is little data between 0.05 and 1 dpa and current data indicates some toughness decrease between 0.3 and 1 dpa. Would a reduction in the screening threshold from 1 to 0.3 or 0.05 dpa result in additional components screened in for IE? If so, identify the CASS components that would be screened in for IE susceptibility due to these lower screening thresholds. Finally, many CASS components are in the A inspection category. Provide the basis/justification for placing these components in the A category.

    b. The fracture toughness of CASS can degrade significantly due to thermal embrittlement (TE) and the toughness of both CASS and other stainless steel materials can decrease significantly as neutron fluence increases. When the dose exceeds 5 dpa, available data indicates that fracture toughness can be extremely degraded in many materials. The staff's concern is that the fracture toughness in CASS components may get so low due to TE and/or IE that preexisting fabrication or service-induced flaws that are smaller than the inspection resolution may challenge component integrity under normal loading or under design basis events. Additional guidance to licensees may be needed either in MRP-227 or WCAP-17096, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," Revision 2, December 2009, to address this situation. Describe how existing or planned guidance addresses this issue. Otherwise, justify why such guidance is not needed.

16. MRP-227 identifies several components that require plant-specific aging analysis (e.g., fatigue analyses) to determine the appropriate inspection category. However, MRP-227 does not discuss or reference approved methods or acceptance criteria for conducting such analyses. Discuss why guidance is not necessary to ensure consistent application and interpretation of plant-specific aging analyses. Alternatively, if the industry plans to provide such guidance, discuss the plans, approach, and schedule for developing this guidance. This discussion should address how environmental effects should be treated in these analyses.

17. MRP-227 Tables 3-1, 3-2, and 3-3 identify component/aging mechanism combinations (where the combination is identified as either "primary," "expansion," or "covered by existing programs" in the tables) that are not identified in Tables 4-1 through 4-6, 4-8 and 4-9. Tables 3-2 and 3-3 also identify components (e.g., In-Core Instrumentation Thimble Tubes in CE internals and control rod guide tube support pins in Westinghouse Internals) that are not identified in Tables 4-1 through 4-6, 4-8 and 4-9 at all. Identify the component/aging mechanism combination in Tables 3-1, 3-2, and 3-3 that are not identified in Tables 4-1 through 4-6, 4-8 and 4-9. Explain how each of these aging mechanisms will be managed and revise MRP-227, if appropriate. If the aging management review (AMR) line items that were previously provided to update the Generic Aging Lessons Learned Report in conjunction with the staff's review of MRP-227 need to be revised, provide recommended changes to the AMR line items.

18. As a follow-up to RAI 2-26, 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.

19. The following components, listed as an example only, were originally identified for potential aging degradation but they were dispositioned under "No Measures" category. Provide an explanation for not performing any analysis prior to binning them under the "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

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.

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.

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 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.

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.

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 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.

a) Combustion Engineering---Core Support Column Bolts; Core Shroud Bolts; Guide Lug Insert Bolts; Barrel-Core Shroud Bolts

b) Westinghouse----Baffle-edge Bolts; Baffle-former Bolts; Lower Support Column Bolts

c) 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.

25. The effects 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.

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 the PWR vessel internals program.

26. RAI 2-20 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 2-20 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 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 significant degradation is not occurring in the expansion components such that the design margin requirements for expansion components and associated systems are satisfied.

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. Provide an overview of how these issues were evaluated to determine the final AMP recommendations for welded components and provide specific examples to illustrate the impact of these issues on the final inspection requirements.

EPRI | ELECTRIC POWER RESEARCH INSTITUTE

MRP Materials Reliability Program________________________MRP 2010-066
(via email)

October 29, 2010

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

Subject: EPRI MRP Final Response to: REQUEST FOR ADDITIONAL INFORMATION NUMBER 4, RE: ELECTRIC POWER RESEARCH INSTITUTE TOPICAL REPORT 1016596,'MATERIALS RELIABILITY PROGRAM: PRESSURIZED WATER REACTOR INTERNALS INSPECTION AND EVALUATION GUIDELINES (MRP-227-REV. 0)' (TAC NO. ME0680), August 30, 2010

To Whom It May Concern:

Enclosed are two copies of the subject document provided in PDF format on compact disc (CD) storage devices. This material is non-proprietary.

If you have any questions on this item, please contact Anne Demma (ademma@epri.com) at 650-855-2026.

Sincerely,

Terry McAlister
SCANA
Chairman, Materials Reliability Program

Cc: James Lash, First Energy
Sheldon Stuchell, NRC (with 8 copies of Subject document)
Victoria Anderson, NEI
David Steininger, EPRI
Chuck Welty, EPRI

Final
Responses to 4th Set RAIs
On MRP-227, Rev 0

October 29, 2010

MRP-227 Request for Additional Information (RAI) #4

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

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

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

RAI 4-1: Develop a comprehensive roadmap that describes how components that were categorized as non-A (i.e., Category B and C components) by the initial screening analysis were binned into the final recommended inspection categories (i.e., primary, expansion, existing, and no additional measures). The roadmap should include, for a sufficient number of components to demonstrate the general practice, the results of the initial screening analysis, the failure modes, effects, and criticality analysis (FMECA) susceptibility and consequence results and rationale supporting the results, a brief summary of the functionality assessment results (if applicable) and a description of the use of these results (i.e., impact) in defining the recommended inspection program, a summary of the recommended inspection program (i.e., inspection type, periodicity, and accessibility requirements), and the supporting basis for this program. As an integral part of this demonstration, ensure that justification/rationale exists for classifying initial B and C components with medium or high failure consequences in any bin other than the primary category. The roadmap should cite references and data sources used to develop information/justification supporting the recommended ranking for each component discussed (e.g., consequence analysis).

The roadmap should also identify, for each component discussed, the loading sources that provide the normal operating stresses considered for each component. Loading sources may include, for example, pressure, thermal, deadweight, residual stress (e.g., from fabrication/installation, welding), hydrodynamic, and preload stresses. The roadmap should identify which, if any, of these stresses may produce significant cyclic or transitory stresses under normal operating conditions. Please indicate the portion of the normal operating stress due to static loading sources and the portion attributed to significant cyclic or transitory load sources that may contribute to fatigue.

Response: A separate roadmap document has been developed to augment the RAI responses and is included as part of the overall response package as Appendix B. It describes the eight-step process from component identification through assignment of inspection recommendations. It also points to the MRP documents which contain the details of the evaluations performed in the development of MRP-227. The roadmap does contain examples of how components were screened in or out, classified and re-evaluated.

The following are examples of the process used for binning components.

Example: Westinghouse Bottom Mounted Instrumentation Column Bodies listed as Expansion Item

Original screening results: MRP-191 Table 5-1

- Mechanisms: SCC, IASCC, Irradiation Embrittlement, Fatigue, Void Swelling Functional Description:
- MRP-232 Section 4.2.6: BMI column assemblies provide a path for the flux thimbles into the core from the bottom of the vessel and protect the flux thimbles during operation of the reactor.

FMECA Conclusion: MRP-191 Table 6-5

- Medium Failure Probability, Low Consequence Analysis of Degradation Mechanisms: MRP-232 Section 4.2.6.1
- Expansion based on cracking in CRGT lower flanges
  - The primary function of the BMI columns is to allow insertion and withdrawal of the flux thimbles, and as was noted several times, failures within the columns would be indicated by difficulty with the insertion of the flux thimbles during a refueling outage. Thus, detailed inspections are not required, and this component is classified as being an Expansion inspection component, required only when the regular withdrawal and insertion of the flux thimble indicates malfunction.
  - Analysis of lower core plate indicated irradiation effects are overestimated.
  - BMI system has no structural function.

<u>Example: B&W Core barrel cylinder</u>

The core barrel supports the fuel assemblies, lower grid, flow distributor, and incore instrument guide tubes. The primary function of the core barrel cylinders and welds during normal power operation is to provide a flow envelope for the core and, thereby limit core bypass flow.

The core barrel cylinders and welds therefore do not have a direct core support safety function; however, they do have a safety function to control bypass around the core during a loss-of-coolant-accident (LOCA).

Initially screened in as <u>Non-A</u> and ultimately grouped as <u>Expansion</u>

- Screened in as <u>Non-A</u> for SCC, fatigue, and irradiation embrittlement in Step 3 (austenitic stainless steel, Type 304 with welds), all other mechanisms screened out

- FMECA expert panel determined that fatigue as an aging mechanism to have a low susceptibility that is supported by no known operating experience of fatigue, and the design criteria containing a significant amount of margin

- FMECA results identified SCC susceptibility as "B" and safety consequences as "1," which preliminarily categorizes this item as "Category A"

- FMECA results identified IE susceptibility as "C" and safety consequences as "1," which preliminarily categorizes this item as "Category B" (see table in Step 5)

- As shown in Section 3.2.3 (MRP-231) the core barrel cylinder is considered inaccessible and is not part of the standard 10-year ISI inspection. However, limited access to the former plates, core barrel cylinder, and otherwise inaccessible bolt locking devices is available through the flow bypass holes should a limited examination become necessary

- The baffle plates are the primary item for inspection from IE while the core barrel cylinder is considered to be Expansion item due to its low safety consequences and lower dose

Regarding the general treatment of loads; specific loads on individual components were not explicitly considered. However, design loading for normal operating conditions (e.g. pressure, thermal, dead-weight, residual stress, etc.) as well as design-basis loads were considered on the basis of relative magnitude and impact on operability and functionality using engineering judgment through the FMECA process. Additional insights into the treatment of loads by AREVA for B&W plants and Westinghouse for the CE and Westinghouse plants follow.

B&W:

Screening of PWR RV internals components was performed using the MRP-175 screening criteria in Table 3-2 of MRP-175. The screening process and results for the B&W RV internals are documented in MRP-189 Rev. 1. Of the eight aging degradation mechanisms, only SCC and IASCC require applied tensile stress value as a screening parameter.

The original qualification of the B&W RV internals for B&W 177-FA was accomplished by both analytical and test methods. Therefore, original stress from the late 1960s and 1970s analyses using ASME Section III as a guidance were found only for a limited number of RV internals components including the stress analyses intended for the B&W 205-FA. In general, the maximum calculated stress values under normal and upset condition were used for screening SCC and IASCC. For non-bolting components, the loading sources for the stress values used included hydraulic pressure loads, dead weight, preload, operating basis earthquake (OBE), and flow-induced vibration (FIV) loads while the thermally-induced load was not included. For bolting components, the loading sources for the stress values used included hydraulic pressure loads, dead weight, preload, OBE, and FIV and thermally-induced loads.

In addition to the earlier stress analyses described above, detailed stress calculations were performed in the 1980s for a number of RV internals high-strength bolts (i.e., Alloy A-286 and Alloy X-750) to address IGSCC identified at the time. The stress values listed in MRP-189 for screening these high-strength bolts were from the stress calculations performed in the 1980s. In these cases, the loading sources included flow, dead weight, preload, and FIV and thermal loads from steady-state and transient operations.

Note, the applied stress defined in MRP-175 includes normal operating stress under steady-state condition and residual stress from fabrication and welding. However, residual stress due to welding or fabrication processes was not considered in any of the above stress calculations. Residual stress due to welding and fabrication processes was only addressed by use of the additional screening parameters "Multiple-Pass Weld" and "Cold-Work > 20%" during the screening process.

Westinghouse:

> Specific loads on individual components were not explicitly considered on a plant by plant or design grouping basis. Typical loadings for normal operating conditions (e.g. pressure, thermal, dead-weight, residual stress, water hammer, etc.) were considered as the primary drivers for manifestation of degradation. This view was based on currently known operating histories. Design-basis loadings and combinations of loadings for the range of as licensed conditions (e.g. normal, upset, emergency, faulted) were considered on the basis of relative magnitude against typical allowable limits as defined from historical plant evaluations. In all cases the loads employed were within the acceptable bounds for various plant conditions based on a variety of as licensed Code requirements. Relative magnitudes varied widely as a result of the variety of as licensed conditions for the individual units when considering seismic and LOCA conditions as well as the range of normal operating transient effects. Where available, fatigue was considered from both a usage factor and range of stress perspective in considering effects. Potential postulated impacts on operability and functionality using engineering judgment through the FMECA process aided in determining levels of concern regarding operational functionality and safety.

RAI 4-2: Develop a comprehensive roadmap for components that were categorized as A by the initial screening analysis and have medium or high failure consequences. This roadmap should include the consequence results and supporting rationale associated with the recommended categorization. Indicate if and why any of these components were moved into the primary or expansion categories based on either the FMECA or functionality analysis. The roadmap should also cite references and data sources used to develop information/justification supporting the recommended ranking for each component (e.g., consequence analysis).

The roadmap should also identify, for each of the medium or high failure consequences component, the loading sources that provide the normal operating stresses considered for each component. Loading sources may include, for example, pressure, thermal, deadweight, residual stress (e.g., from fabrication/installation, welding), hydrodynamic, and preload stresses. The roadmap should identify which, if any, of these stresses may produce significant cyclic or transitory stresses under normal operating conditions. Please indicate the portion of the normal operating stress due to static loading sources and the portion attributed to significant cyclic or transitory load sources that may contribute to fatigue.

Response: A separate roadmap document has been developed to augment the RAI responses and is included as part of the overall response package as Appendix B. It describes the eight-step process from component identification through assignment of inspection recommendations. It also points to the MRP documents which contain the details of the evaluations performed in the development of MRP-227. The roadmap does contain examples of how components were screened in or out, classified and re-evaluated.

Using the Issue Management Tables (MRP-156 and MRP-157) the failure consequences are defined as:

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

For those components placed in Category A, all but one component (Lower Support Casting or Forging) the only failure consequence identified was "G, Causes a significant economic impact". As noted in Table 2 of the roadmap the Lower Support Casting or Forging also had a consequence "A, Precludes the ability to reach safe shutdown" assigned to it. As is noted in the roadmap section for the FMECA process:

> "…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. However, the FMECA panel could not identify any potential cause or mode of catastrophic failure that would require aging management of this large forging. Therefore the lower support forging was not reinstated for additional evaluation.

Also, as part of the FMECA all components initial screened as "A" components were reevaluated by the FMECA panels. In each case the "A" classification was confirmed.

Regarding the general treatment of loads; specific loads on individual components were not explicitly considered. However, design loading for normal operating conditions (e.g. pressure, thermal, dead-weight, residual stress, etc.) as well as design-basis loads were considered on the basis of relative magnitude and impact on operability and functionality using engineering judgment through the FMECA process. Additional insights into the treatment of loads by AREVA for B&W plants and Westinghouse for the CE and Westinghouse plants follow.

B&W:

Screening of PWR RV internals components was performed using the MRP-175 screening criteria in Table 3-2 of MRP-175. The screening process and results for the B&W RV internals are documented in MRP-189 Rev. 1. Of the eight aging degradation mechanisms, only SCC and IASCC require applied tensile stress value as a screening parameter.

The original qualification of the B&W RV internals for B&W 177-FA was accomplished by both analytical and test methods. Therefore, original stress from the late 1960s and 1970s analyses using ASME Section III as a guidance were found only for a limited number of RV internals components including the stress analyses intended for the B&W 205-FA. In general, the maximum calculated stress values under normal and upset condition were used for screening SCC and IASCC. For non-bolting components, the loading sources for the stress values used included hydraulic pressure loads, dead weight, preload, operating basis earthquake (OBE), and flow-induced vibration (FIV) loads while the thermally-induced load was not included. For bolting components, the loading sources for the stress values used included hydraulic pressure loads, dead weight, preload, OBE, and FIV and thermally-induced loads.

In addition to the earlier stress analyses described above, detailed stress calculations were performed in the 1980s for a number of RV internals high-strength bolts (i.e., Alloy A-286 and Alloy X-750) to address IGSCC identified at the time. The stress values listed in

MRP-189 for screening these high-strength bolts were from the stress calculations performed in the 1980s. In these cases, the loading sources included flow, dead weight, preload, and FIV and thermal loads from steady-state and transient operations.

Note, the applied stress defined in MRP-175 includes normal operating stress under steady-state condition and residual stress from fabrication and welding. However, residual stress due to welding or fabrication processes was not considered in any of the above stress calculations. Residual stress due to welding and fabrication processes was only addressed by use of the additional screening parameters "Multiple-Pass Weld" and "Cold-Work > 20%" during the screening process.

Westinghouse:

Specific loads on individual components were not explicitly considered on a plant by plant or design grouping basis. Typical loadings for normal operating conditions (e.g. pressure, thermal, dead-weight, residual stress, water hammer, etc.) were considered as the primary drivers for manifestation of degradation. This view was based on currently known operating histories. Design-basis loadings and combinations of loadings for the range of as licensed conditions (e.g. normal, upset, emergency, faulted) were considered on the basis of relative magnitude against typical allowable limits as defined from historical plant evaluations. In all cases the loads employed were within the acceptable bounds for various plant conditions based on a variety of as licensed Code requirements. Relative magnitudes varied widely as a result of the variety of as licensed conditions for the individual units when considering seismic and LOCA conditions as well as the range of normal operating transient effects. Where available, fatigue was considered from both a usage factor and range of stress perspective in considering effects. Potential postulated impacts on operability and functionality using engineering judgment through the FMECA process aided in determining levels of concern regarding operational functionality and safety.

RAI 4-3: In addition to the information supplied in response to Questions 1 and 2, the FMECA process should be more fully documented to support its use in the development of the recommended aging management programs. Discuss and describe the following aspects of the FMECA analysis for both the Westinghouse/CE and B&W studies[1]:

a. Relate the list of experts who participated in the FMECA with the list of required technical specialties to perform this analysis.

[1] Note that some of this information was presented during the meeting on June 8, 2010 between the industry and the NRC.

b. Describe the FMECA process used in each study including the following items:

- key assumptions,
- scope and motivation (e.g., what components addressed, use to confirm initial screening results, use to develop recommendations for component classification),
- approach (e.g., ranking and consensus process used to obtain results, consideration of either single degradation mechanisms or combined mechanisms, ranking definitions, development of classification matrices, use of classification matrices to development component classification recommendations, evaluation of the effects and consequences of component failure for Westinghouse, CE, and B&W designs), and
- analysis and results (e.g., how individual estimates were amalgamated to determine final estimate, how ranking biases among the various experts were addressed and reconciled, results of both susceptibility and consequence analyses, the impact of the FMECA on the final severity of component failure rankings with a focus on instances where the FMECA was used to change initial severity rankings).

In particular, identify and then discuss differences between the Westinghouse/CE and B&W FMECA studies. This discussion should address, for example, differences in the susceptibility/failure likelihood and severity/damage likelihood definitions, consideration of the effects of either single or multiple degradation mechanisms, recommended inspection categories based on the tabulated matrix values,. Additionally, this discussion should, identify any Westinghouse, CE, and B&W components having a similar or equivalent function that received different aging management program classifications based on the FMECA, and, as appropriate, either document why the differences exist or describe how differences in the Westinghouse/CE and B&W FMECA results for similar components were reconciled.

c. One of the FMECA assumptions states that "...no consideration was given to manufacturing errors, maintenance errors, installation errors, transport errors, or any other type of random or human errors." Why aren't these failure modes considered for components that have complex manufacturing, maintenance, or installation procedures? What is the justification for not considering these effects for components with medium or severe failure consequences?

d. Section 6.2 or MRP-191 indicates that a "...FMECA does not serve well to identify multiple failures." What is meant by multiple failures (i.e., common cause, indirect, cascading failures, or another type)? How is this deficiency in the FMECA process addressed as part of the aging management strategy?

e. Section 6.2 or MRP-191 also indicates that "…operability, reliability, and availability issues were also considered." This statement is unclear. What explicit issues were considered in the FMECA and how were they considered by the experts? For example, was there an explicit process to factor these issues into the FMECA rankings or were these issues used to determine which of several components with similar degradation mechanisms and likelihood and/or failure consequence would be chosen for the primary inspection category?

Response: The response to the various parts of this RAI will take several forms. In some cases, the response to a particular part will point to a section in either MRP-190 or MRP-191 that directly addresses the issue. In other cases, the response has already been provided by the industry in a presentation to the NRC staff during the June 8, 2010, meeting between the industry and the NRC staff, and the response will so indicate. Finally, in some cases, the response is available in the separate roadmap document that has been developed to augment the RAI responses. That separate roadmap document is included as a part of the overall response package.

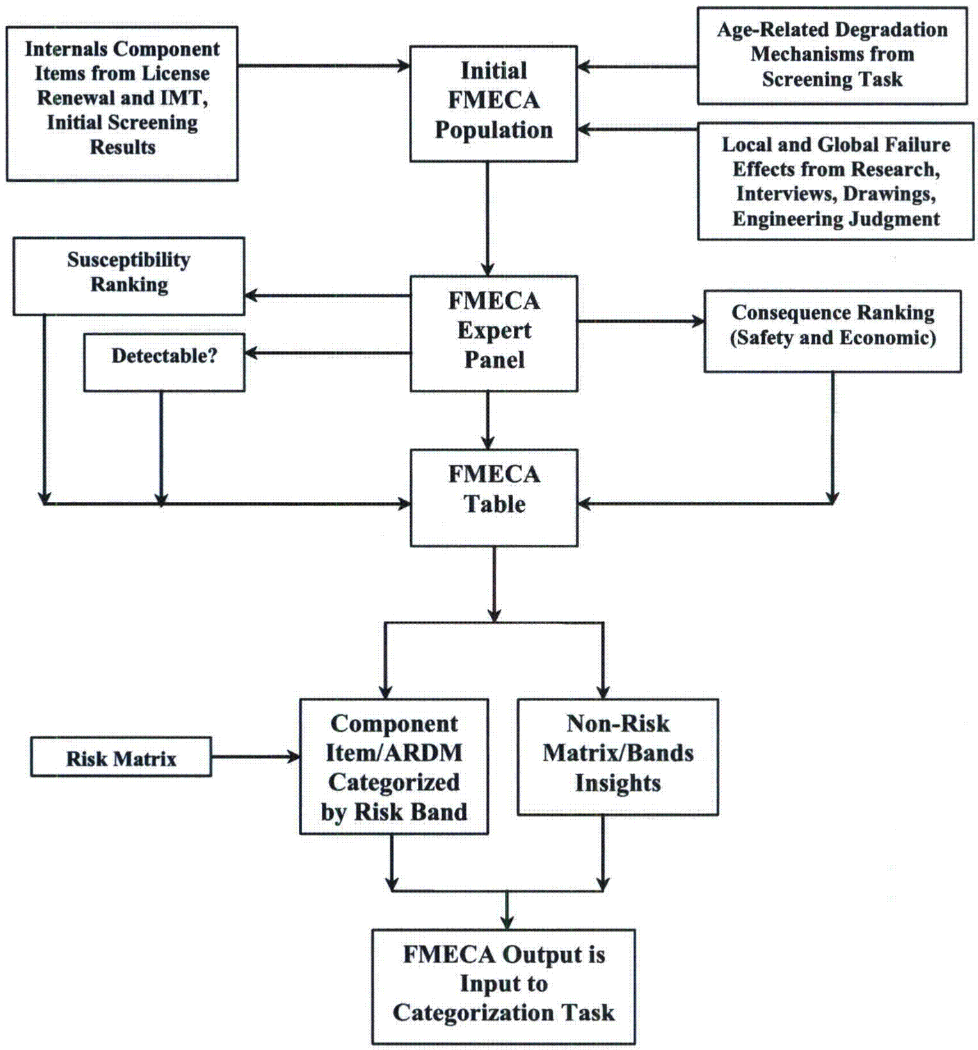
RAI 4-3(a) asks for the list of experts who participated in the FMECA process and their technical areas of expertise. The list of experts for the FMECA documented in MRP-190 and their areas of expertise is shown on the Acknowledgements on Page ix of MRP-190. MRP-191 does not provide the list of experts and their areas of expertise, but the areas of expertise were provided in the presentations given to the staff on June 8, 2010. For completeness, the Combustion Engineering/Westinghouse FMECA areas of expertise are shown below:

- Component design, testing and repair
- Structural modeling and analysis
- Thermal-hydraulics and systems analysis
- Neutron fluence and radiation analysis
- Materials degradation and failure experience
- Component inspection experience
- Risk assessment
- Inspection requirements
- System function and operating experience
- Licensing and regulatory interaction

The list of experts and areas of expertise acknowledged in MRP-190 explicitly called out eighteen people -- designers, materials experts, radiation physics and neutronics/safety experts, thermal hydraulics systems engineers, stress and structural analysts, non-destructive examination experts, safety and accident analysts, license renewal (aging management) specialists, and project managers. While not specifically cited, these individuals in most cases were also experienced in component fabrication, installation, and operation. For completeness, the B&W FMECA areas of expertise are shown below:

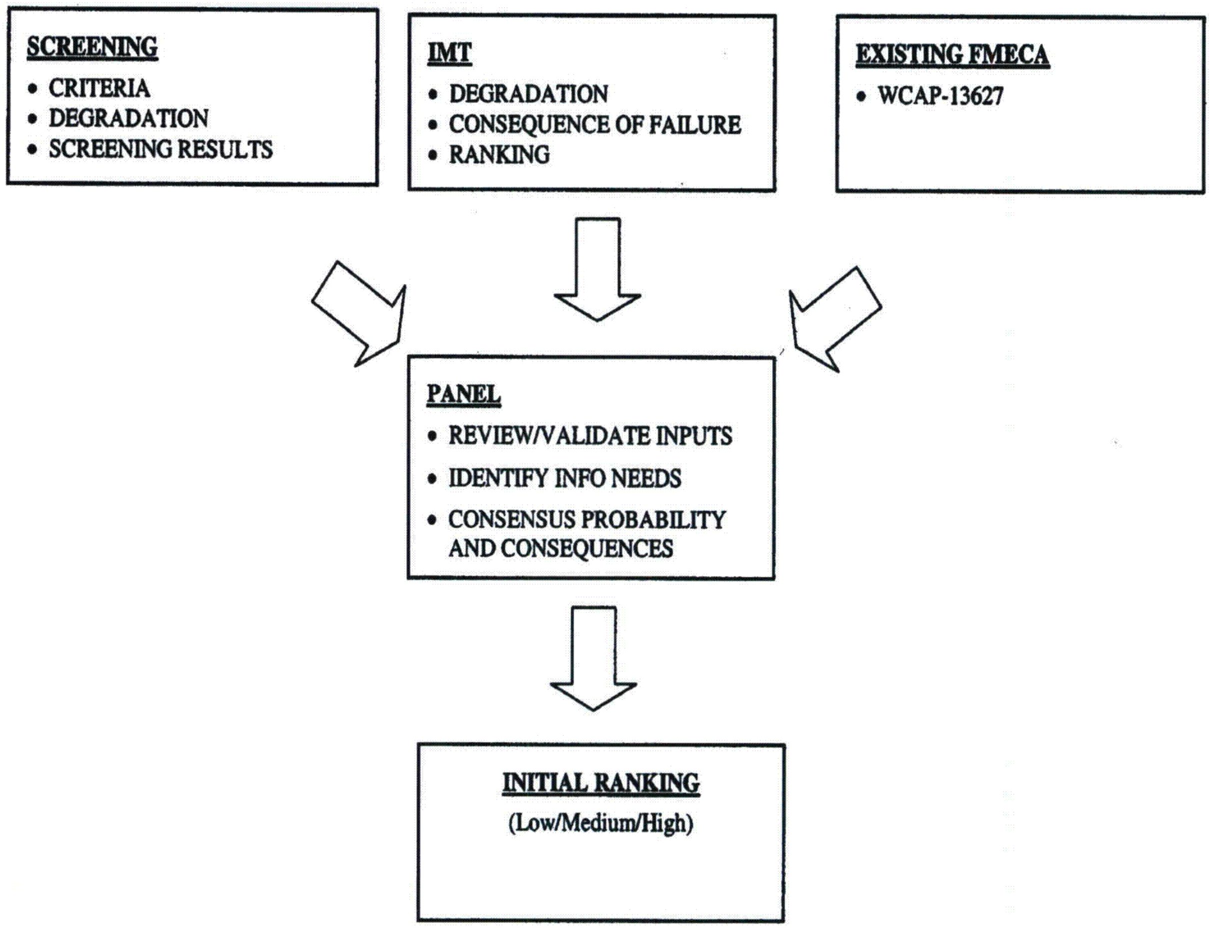
- Component Design
- Materials
- Finite Elements/Stress/Dynamics Analysis
- Structural Analysis
- Non-Destructive Evaluation
- Fracture Mechanics
- License Renewal
- Safety/Accident Analysis
- Thermal-Hydraulics
- Neutronics/Radiation Physics

RAI 4-3(b) essentially asks for a description of the FMECA process used by each vendor, with any substantive differences in process identified. For MRP-190, most of this type of information is described in Section 3.1, but additional information on the process is given in other portions of Section 3 and the key assumptions are provided in Section 4. While the details of the FMECA deliberations were not recorded, the FMECA process steps can be followed reasonably well by starting with the flow chart shown below, and then using the table format in Appendix A to follow the logical exercise of that flow chart.



The tables in Appendix A of MRP-190 contain eleven columns, starting with the component name, the FMECA identifier, the degradation mechanism(s), the failure mode(s), and the local and global failure effects. The next three columns reflect the deliberations of the experts after developing the failure mode(s) and failure effect(s) columns, giving the consensus ranking of susceptibility and the safety and economic consequence severity. The transition from the former columns to the latter columns represents the aggregate deliberations of the expert panel. The final two columns – the determination of detectability and the column with comments include insights representing the collective informed judgment of the experts, and are invaluable in understanding how the FMECA process results evolved.

For MRP-191, the description of the Combustion Engineering/Westinghouse FMECA process is largely provided in Section 6.2, and the flow chart of that process is shown below.



The FMECA expert panel that followed this flow chart process was a multi-disciplinary team assembled to consider failure modes, effects and consequences. Some individuals from the screening panel were also included in the expert panel. However, reactor internals design and analysis was only one of the many disciplines represented on the FMECA panel. The entire FMECA panel was briefed on the screening process as part of their training and data generated by the screening panel was also made available to the FMECA panel. The FMECA panel was asked to review and confirm the results of the screening process.

To supplement the screening process, panels of experienced stress analysts and designers were assembled to augment the available information with conservative estimates of the required values. Separate panels were created for the Westinghouse and CE reactor designs. Each panel was comprised of individuals familiar with the original design basis for the internals and the range of design variants. The screening panels were instructed to generate generic answers to six basic questions:

1. Could the operating stress be >30 ksi?
2. Where is the component located relative to the core?
3. Is there potential for wear?

4. Could the Cumulative Fatigue Usage Factor (CUF) be >0.25 @ 40 years?
5. Does it contain a structural weld?
6. Is the component bolted or is it a spring?

The questions were structured to conservatively assure compliance with the requirements of the MRP-175 screening process. Whenever there was a potential for a component to exceed the stated limit, the panel answered the question in the affirmative. The role of the screening panel was limited to generating input data for the screening process.

RAI 4-3(b) also asked about any substantive differences in the two FMECA processes. The major difference between the two approaches is that the screening and categorization was documented in a separate report (MRP-189) for the B&W plants, while that portion of the work was combined into a single report (along with the FMECA results) in MRP-191 for both CE and Westinghouse plants. The style of reporting the FMECA results is quite different, as well. The summary tables in MRP-190, Appendix A (Tables A-1 through A-4) give a wealth of information that was derived from the FMECA process, and might be considered to contain much greater detail than is provided in Tables 6-5 and 6-6 of MRP-191. The latter tables summarize the results of the FMECA process for CE and Westinghouse internals without the detail of expert elicitation that can be found in Tables A-1 through A-4 of MRP-190. Additional detail on the FMECA process deliberations of MRP-191 was provided in the presentations given on June 8, 2010, to the NRC staff, and further discussion is provided in the accompanying roadmap document. However, at the summary level required for decision-making and incorporation into MRP-227, any differences between the two approaches are not significant.

Should additional information on the expert panel deliberations that led to the FMECA results cited in Tables 6-5 and 6-6 of MRP-191, it will be necessary to meet with some of the cognizant individuals involved in the Combustion Engineering/Westinghouse FMECA deliberations under conditions that protect the proprietary nature of the discussions that eventually led to the cited results.

RAI 4-3(c) asks about a statement in Section 6.2 of MRP-191 that "…no consideration was given to manufacturing errors, maintenance errors, installation errors, transport errors, or any other type of random or human errors." This statement is consistent with the purpose of MRP-227, which is the management of age-related degradation effects, with such effects as installation errors handled by other programmatic efforts, such as the plant corrective action program. This statement is also consistent with the guidance provided in Branch Technical Position RSLB-1 (see pages A.1-1 through A.1-9 of NUREG-1800, Revision 2).

RAI 4-3(d) asks about a statement in Section 6.2 of MRP-191 that "…FMECA does not serve well to identify multiple failures." This statement means that the FMECA process provided a limiting level of protection at the initiating event stage and thus did not address hypothetical multiple failures.

RAI 4-3(e) asks about a statement in Section 6.2 of MRP-191 that "...operability, reliability, and availability issues were also considered." The purpose of this statement was to point out that the expert elicitation team for the FMECA process included experts on internals component operability, reliability, and availability. So the answer is yes, the FMECA process includes explicit consideration of these issues by the experts, and those expert opinions were taken into consideration in the development of the inspection recommendations.

RAI 4-4: In addition to the information provided in response to Question 1, discuss the impact of the functionality analysis in terms of determining or modifying the final inspection requirements for components. Specifically, provide an overview of how the functionality analysis impacted the recommended component classifications (i.e., primary, expansion, existing programs, no additional measures) that were initially developed prior to conducting the functionality analysis. Also, for primary and expansion components indicate how the functionality analysis was used to determine the type of inspection and the inspection periodicity. Finally, identify all similar Westinghouse, CE, and B&W components (i.e., those that perform a similar function and have similar failure consequences) that have different final inspection classification and requirements based on the functionality analysis. Provide justification for any differences that exist.

Response: A separate roadmap document has been developed to augment the RAI responses and is included as part of the overall response package as Appendix B. It describes the eight-step process from component identification through assignment of inspection recommendations. It also points to the MRP documents which contain the details of the evaluations performed in the development of MRP-227. The roadmap does contain examples of how components were screened in or out, classified and re-evaluated.

For the specific overview requested in this RAI, refer to steps 5, 6 and 7 of the roadmap and the MRP document text references identified in those steps.

Components, whether they perform similar functions or not, or have similar failure consequences were evaluated on their own based on multiple criteria. A decision was then made on where it should be binned for inspection. The following examples shown below depict similar components with different classifications..

1. Upper core support barrel flange weld (W & CE) vs. CSS upper flange weld (B&W)

The upper core support barrel flange weld in W and CE units is "Primary" due to SCC. The equivalent weld location in B&W units is the core support shield (CSS) upper flange weld, which is Category "A" and does not require augmented aging management.

The CSS upper flange weld in B&W units was screened out of all aging degradation mechanisms including SCC as documented in MRP-189 Rev. 1, Tables 3-2 and 3-3 with the FMECA identifier "S.2". This weld is a double U-groove weld made using an automatic submerged arc welding process with Type 308L weld metal. The MRP-175 SCC screening criteria for austenitic stainless welds are 30 ksi and <5% ferrite. Due to the low applied stress on the CSS upper flange and specified minimum 5% ferrite for austenitic stainless welds by B&W, this weld location was screened out of SCC for the B&W units. In addition, the CSS cylinder received post-weld stress relief treatment after the top and bottom CSS flanges were welded in the shop during the fabrication.

The upper core support barrel flange weld in W and CE units is identified in MRP-232, Table 2-18 (W) and Table 2-19 (CE) with a initial category "B" or "C", and the final group "Primary".

2. CRGT guide cards (W) vs. CRGT guide tubes and sectors (B&W)

   The CRGT guide plates (cards) in W units are "Primary" due to wear. The equivalent components in B&W units are the CRGT guide tubes and guide sectors, which is in the "No Additional Measure" category.

   The CRGT guide tubes (C-tubes) and guide sectors (split-tubes) in B&W units were initially categorized as "Not-A" for Wear, and were placed in Category "B" after the FMECA as documented in MRP-190 and MRP-189 Rev. 1 Table 4-1. Afterwards, wear of the CRGT guide tubes and sectors was further evaluated by reviewing past wear investigations to the control rods within the guide path as documented in MRP-231 Section 2.3. It was concluded that there was no evidence of wear on the control rod, and thus there should not be any wear on the CRGT guide tubes and guide sectors. Therefore, the CRGT guide tubes and sectors were downgraded to the "No Additional Measure" category.

   The designs of the CRGT assembly are quite different for B&W and W units. The guidance for the control rods in B&W units is continuous with full-length guide tubes and sectors. In W units, approximately ¼ of the CRGT assembly length guidance near the bottom is provided using continuous guide tubes (called C-tubes and sheaths in MRP-191 and MRP-232). The guidance for the remaining CRGT assembly length is discontinuous using guide plates (cards). For W-units, the C tubes and sheaths were initially categorized as "C" in MRP-190 and "Primary" in MRP-232 Table 2-6 for wear. However, they were finally placed in the "No Additional Measures" category in MRP-232 Table 2-19. Note 2 to MRP-232 Table 2-19 explained that the C-tubes and sheaths were placed in the "No Additional Measures" because decisions on remediation of wear and degradation in the CRGT assembly will be based only on the conditions detected in the "Primary" the guide tubes (cards).

RAI 4-5: In addition to the information provided in response to Questions 1 and 3, discuss in general how susceptibility to multiple degradation mechanisms was considered when developing the final inspection recommendations. The final inspection category recommendations for each component appear to typically be based on the susceptibility and consequences associated with the single most-dominant degradation mechanism. However, many components are subject to multiple degradation mechanisms and it is not clear how the synergistic effect of multiple degradation mechanisms was considered in the final recommendations. The concern is that a component subjected to multiple degradation mechanisms may be more likely to experience a greater level of total degradation than a component that is subject to a single mechanism (even though the component with the single mechanism may be more susceptible to that mechanism). Also discuss the acceptability of the recommended inspection method for primary components that are susceptible to multiple degradation mechanisms. Demonstrate that the recommended method is capable of identifying degradation due to all significant contributing mechanisms (and not just the single most-dominant mechanism) before component design margins are exceeded.

Response: Susceptibility to multiple degradation mechanisms was considered during the initial screening process, as documented in MRP-189 for B&W internals components and in MRP-191 for both Combustion Engineering and Westinghouse internals components. All internals components were evaluated against the screening criteria for each of the eight identified degradation mechanisms separately. However, in some cases, combinations of degradation mechanisms or effects were known to interact, or had the potential for interaction. The most common of these was the case of irradiation stress relaxation/creep combined with wear (loss of material due to wear) and/or fatigue (cracking due to fatigue) for some bolting. Examples of this are shown in Table 5-1 of MRP-191, where Wear (I) and Fatigue (I) are identified for upper column assembly bolting. The parenthetical (I) is intended to illustrate the effect of irradiation that can lead to potential loss of preload from irradiation-induced stress relaxation/creep, which in turn has the potential to cause loss of material due to wear and/or cracking due to fatigue.

While not directly combined, many components were found to be potentially affected by moderate or significant degradation from more than a single mechanism. Examples of multiple degradation mechanisms with moderate or significant effects can be found in the tables in Section 3 of MRP-227.

The results of the screening process were revisited during the expert elicitation FMECA process, which required a confirmation by the experts that the screening results, including those cases of combined effects, were not contradicted by experience. In addition, the FMECA process included consideration by the experts of all of the degradation mechanisms for which moderate or significant effects were suspected, and these experts were capable of evaluating the potential for combinations of degradation mechanisms to cause more harmful effects than might be caused by individual degradation mechanisms.

Therefore, for many components subject to more than one degradation mechanism with moderate or significant effects, the final inspection category recommendation reflected the need for the inspection to detect an effect common to more than one degradation mechanism (e.g., cracking caused by IASCC and fatigue). And, in a few cases, the final inspection category recommendation reflected the need for an inspection capable of detecting more than one effect during the same examination (e.g., distortion caused by void swelling; gross cracking and material separation caused by IASCC). For this particular case, a visual (VT-3) examination encompasses the relevant conditions that describe both distortion caused by void swelling and material separation caused by gross cracking.

In summary, potential susceptibility to the effects from multiple degradation mechanisms was considered by: (1) identifying such combinations during the initial screening based on known interactions (e.g., irradiation-induced stress relaxation of bolt pre-load combined with either wear or fatigue); (2) FMECA expert elicitation of combined effects that resulted in greater consequences; and (3) recommending examinations capable of detecting relevant conditions caused by more than one degradation mechanism or effect.

RAI 4-6: Several previous RAIs (e.g., RAIs 11 and 18 (Set #2), and RAI 3-8) have questioned whether plant-specific analyses are required to demonstrate that each plant is appropriately represented by MRP-227 such that the proposed aging management programs (AMPs) are applicable. That is, confirmation that the plant's initial design and operating conditions fall within the scope of the MRP-227 evaluation, the plant complies with important assumptions underlying the MRP-227 analysis, and changes in plant design or operating conditions (e.g., resulting from power uprates) have been appropriately considered. Meeting these conditions is necessary to ensure that the plant-specific AMP inspection requirements (i.e., the primary inspection components, inspection type and periodicity) would not be different from the MRP-227 recommendations determined through more generic evaluation.

The responses to these various RAIs have indicated that a plant-specific analysis to demonstrate the applicability of MRP-227 guidance is not required because plant-specific differences have been considered by: (a) evaluating operating experience throughout the commercial fleet; (b) using a conservative "out-in" core loading pattern in the functionality analysis; and, (c) assessing several known plant-specific conditions in the FMECA. The responses also justify the representativeness of MRP-227 because (a) base load operational profiles (i.e., fixed power levels) are similar among plants, and (b) no design changes have been enacted by plants other than those identified in generic industry guidance or recommended by the original nuclear steam system supply vendors. However, given the variability in design and operational conditions that currently exists in PWR plants, the staff is not convinced that that the MRP-227 AMP requirements are necessarily appropriate for each plant. For instance, it is not clear that the "out-in" core loading pattern is conservative given that some degradation mechanisms do not initiate until low-leakage core conditions are imposed in the functionality model.

Therefore, the staff requests that guidance be developed that will allow individual licensees to assess the applicability of the MRP-227 method and results. This guidance should particularly focus on demonstrating the applicability of (a) the FMECA and functionality assessments, and (b) the recommended inspection category, inspection method and periodicity for each component. Specifically, this guidance should allow a licensee to determine if plant-specific differences in the RVI design or operating conditions (i.e., power uprate level) result in different component inspection categories (i.e., primary, expansion, existing, and no additional measures) than recommended within MRP-227. Alternatively, additional analysis or justification may be provided to demonstrate that the MRP-227 approach and results are generically applicable such that plant-specific differences in the RVI design or operating conditions do not result in different component inspection categories then recommended within MRP-227.

In that absence of adequate guidance, the staff will consider the need to implement limitations and conditions on the use of MRP-227 which would address plant-specific action items necessary to address this issue for each facility.

Response:
The starting point for the response to this RAI is from Section 2.4 (Guidelines Applicability) of MRP-227, which have been cited in previous RAI responses – The last two paragraphs state that:

"These assumptions are a conservative representation of U.S. PWR operating plants, all of which implemented low leakage core loading patterns early in operating life. The recommendations are thus applicable to all U.S. PWR operating plants as of May 2007 for the three designs identified. These guidelines are also considered applicable to plants that have replaced components or component assemblies; however, alternatives can be technically justified.

Plant modifications made or considered after this date should be reviewed to assess impacts on strategies contained in these guidelines."

These two paragraphs are based on the review and assessment by vendors that: (1) even though power uprates were not specifically addressed in the representative plant component functionality analyses, all plant uprates and other plant modifications up until May 2007 were considered to be within the envelope of the representative plant analysis results; and (2) no inspection recommendation cited in MRP-231 and MRP-232 would have been altered by a change in the functionality assumption of an earlier conversion over to a low-leakage core loading pattern. The first of these vendor findings is covered by the last paragraph, which clearly states the action required by a plant that has sought a power uprate or has undergone a significant plant modification as of May 2007. No further guidance is needed on the power uprate or major plant modification issue. The second of the vendor findings is not intended to argue that degradation mechanisms only initiate during high-leakage core loading operation, or that degradation effects cannot worsen during low-leakage core loading operation. The finding is simply that the vendors have reviewed the functionality analysis results and have determined that the recommended inspection requirements would not be altered by a change in functionality analysis assumption to an earlier conversion from high-leakage to low-leakage core loading. This core loading assumption only has relevance for those components which were aged and assessed using the detailed irradiation analysis finite element analysis (FEA) model. The aging analyses were conducted to understand the complex interactions between active degradation mechanisms in highly irradiated components. These detailed modeling efforts were applied to the B&W and Westinghouse baffle-former-barrel structures, and a welded CE core shroud assembly. The intent of the irradiation aging analysis was to identify trends and limits in the component behavior. The analysis was used to identify factors that could potentially cause component failure. Representative plant designs with relatively severe irradiation conditions were selected for the irradiated aging analysis. These conditions were chosen to test the capability of the structure and identify points of potential concern. While the results of the FEA work provided insights into where degradation would most be expected, neither the vendors nor other members of the core writing team pinpointed the recommended examination scope solely based on the results of irradiation aging analysis. Instead, the irradiation aging analysis results were combined with engineering judgment and experience to provide examination recommendations. The only limited scope recommendations confirmed by the FEA results were for the CE welded designs where the most highly stressed and irradiated weld seams are specified. Thus, while another damage mechanism could play a more important role in the overall aging of the components when a more realistic core loading history is employed, in no case would the recommendations to detect that degradation change because:

- the anticipated effect and the overall degradation hierarchy would not change and
- no component or component assemblies have inspection requirements directed at local effects predicted in the FEA results to the extent that a shift in degradation

mechanism predominance would necessitate a change in location recommendations.

An excellent example of this is provided by baffle-to-former bolts in B&W and Westinghouse plant designs, where the effects of irradiation-induced stress relaxation of bolt pre-load has been observed to reduce the potential susceptibility to IASCC for the baffle-to-former bolt with the highest radiation exposure, while a baffle-to-former bolt with somewhat lower radiation exposure (somewhat further from the core) would be more susceptible. This shift of susceptibility to baffle-to-former bolts further from the core does not lead to a condition where core barrel-to-former bolts are more susceptible to IASCC than baffle-to-former bolts and, since the examination recommendation is for examination of 100% of the accessible baffle-to-former bolts, no change in the examination recommendation is warranted. Therefore, the additional level of detail provided by the functionality analysis does identify complex structural interactions, particularly in bolted assemblies, but did not lead to recommendations for changing the scope of examinations.

While the MRP agrees with the staff that a wide variety of designs are addressed by the representative plants particularly the Westinghouse and CE designs, the plant designs selected do correspond more closely with plants with earlier implementation dates for the MRP-227 requirements.

In addition, as plants begin the implementation process for Issue Program (IP) guidance, such as implementation of MRP-227 guidance, the responsibility for reviewing and determining the applicability of the explicit assumptions given in Section 2.4 are well understood, as outlined in NEI 03-08, including either the need or the wish to use valid alternatives through the deviation process. Thus it would be a plant-specific action to confirm the guidance in MRP-227 remains applicable within the boundaries indicated in Section 2.4. The general framework for both the determination of applicability and the process for justifying deviations is described in industry documents and is further discussed in the response to RAI 4-7.

RAI 4-7: Provide guidance on the process that should be followed by licensees if the plant-specific application of the MRP-227 guidelines identifies that inspection or aging management of a primary component (i.e., as defined in MRP-227) is not necessary. The guidance should address, for example, the plant-specific criteria and process for reclassifying the aging management program for a primary component, disposition of linked expansion components, and identification of (an) alternative plant-specific primary component(s) to be used in lieu of the generic MRP-227 recommendation for that degradation mechanism.

The response to this question should specifically propose text that would be added to the "-A" version of MRP-227 to address this issue.

Response: The plant-specific guidance for implementation of the aging management program elements documented in Tables 4-1 through 4-9 of MRP-227 have been determined to be "Needed" under the Materials Initiative (NEI-03-08). Since MRP-227 guidance has been issued generically for the fleet it is anticipated that utilities may not be able to wholly meet the guidance or may choose to take alternative actions that better correspond with their long term aging management plans, e.g. replacement. Any MRP utility member or contractor personnel to an MRP utility may submit an inquiry to the MRP Advisory Panel of the guidelines for interpretation of part of these guidelines' requirements. In the case of MRP-227, the Advisory Panel core members who serve as the advisory panel on inquiries for MRP-227 were chosen to be the same as the utility members from the guidelines core group who wrote MRP-227-Rev. 0. According to the MRP administrative procedures, the responsibilities of the Advisory Panel are to 1) provide responses to inquiries on the meaning of current MRP Guideline Documents and how they should best be interpreted in light of NEI 03-08 and its other referenced documents; it is not the purpose of the Advisory Panel process to develop new guideline requirements, but to interpret, when necessary, existing requirements documented in the guidelines; and 2) offer recommendations to the guidelines committees regarding revisions to specific guidelines reflecting issues that emerge from the Advisory Panel process. The typical Advisory Panel consideration of an inquiry consists of a telephone conference call or meeting. Advisory Panel decisions are determined by vote, based on a simple majority of the members. Typically, the utility member who filed the inquiry is invited to listen to the Advisory Panel's discussions and respond to questions as needed.

The example of the Inquiry Format (submission via email) is provided below.

Advisory Panel-MRP-### – Topic of Inquiry. Subject:

Point of contact familiar with the reason and basis for inquiry. Name/Phone:

Plant/Utility: Plant Name Unit #/Corporate Name

Section: Section(s) of applicable Guideline

Background: Plant conditions or other circumstances relevant to the inquiry.

Inquiry: Please prepare all statements in a condensed and precise question format. Where appropriate compose inquiry in such a way that "yes" or "no" (perhaps with provisions) would be an appropriate reply.

Proposed Reply: State what it is believed is the intent of the guidelines.

An example of an inquiry to MRP-227-Rev. 0 is provided below:

Inquiry: For plants that do not have a flange of any kind at the location specified in Figure 4-21 of MRP-227, between the continuously guided region and the rest of the lower guide tube, is the intent that we be exempt from this examination category?
Proposed Reply: No, the intent is to examine a sample of locations with high residual and operating stresses to determine that SCC and fatigue are being adequately managed for the internals as a whole. If there is no similar location in a given reactor, then a deviation report with technical bases should be submitted to the MRP.

(Note that the reply to this inquiry was approved by the MRP.)

When a licensee determines that a needed requirement, is not needed, no longer applicable or cannot be accomplished, requirements are already provided in the Implementation Protocol of NEI 03-08. This guidance stipulates that, when a "Mandatory" or "Needed" work product element will not be fully implemented or will not be implemented in a manner consistent with its intent, a technical justification for deviation shall be developed and retained with the utility's program documentation or owner-controlled tracking systems. In addition, deviations from "Mandatory" and "Needed" work product elements shall be entered into corrective action programs (CAP). The technical justification shall provide the basis for determining that the proposed deviation meets the same objective, or level of conservatism exhibited by the original work product, and shall clearly state how long the deviation will be in effect. In the context of MRP-227, if the guidance contained in Table 4-1 through 4-9 can not, need not, or will not be implemented the technical justification for the deviation should clearly state what requirement can not, need not , or will not be met and why; what alternative action is being taken to satisfy the objective or intent of the guidance; and, why the alternative action is acceptable. Examples of alternatives that may be justifiable are: elevation of an Expansion component to Primary; substitution of an equivalent or more rigorous examination than is required by the tables; or destructive testing in lieu of non-destructive examination, such as the case where one or more of the primary components is being replaced. Since the Expansion components are also "needed" requirements, any deviation that would not fully implement a Primary component examination or not implement it in a manner consistent with its intent, would be expected to include in the justification a disposition of associated Expansion components.

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

Although not requested in the RAI, two important steps in the NEI process regarding deviations rest upon the Issue Program (e.g. MRP, BWRVIP, SGMP) that developed the guidance. When a deviation is received, the Issue Program is responsible for reviewing the deviation for technical adequacy, providing feedback to the utility, and assessing the deviation for potential generic applicability and/or need to modify the requirements. Modification to guidance documents can be accomplished via revision or, if warranted, interim guidance can be issued.

This guidance, plus the information contained in MRP-227, its supporting documentation, and in the MRP administrative procedures, is considered sufficient to address the issue raised in the RAI. However, to provide clarity, a paragraph will be added with similar wording to the response above to Section 7.1 of MRP-227 providing a direct reference to NEI 03-08, Implementation Protocol and the deviation justification process (see Appendix A to these RAI responses).

RAI 4-8: This question discusses accessibility requirements for primary and expansion components. Define the appropriate inspection coverage to ensure the component being inspected does not lose its intended function and the process to be followed if the inspection does not meet the inspection coverage. Provide additional guidance on the component accessibility requirements for each primary and expansion component (i.e., those in MRP-227 Tables 4-1 through 4-6, 4-8 and 4-9) such that the results of the inspection can be credited as satisfying the requirements of the aging management program. This guidance should include, at a minimum, the following considerations:

a) For each component, identify the location(s) where degradation is expected.

b) Define the appropriate inspection coverage at this location to ensure that enough of the surrounding material is inspected such that there is assurance that the degradation will be identified before it challenges component or system integrity (i.e., the intended function of the component is retained).

c) Describe the procedure that a licensee should follow if inspection accessibility is insufficient to provide the required inspection coverage or if the inspection does not meet other minimum requirements as specified in MRP-227 and MRP-228.

This procedure must address providing an appropriate justification for continued operation with the reduced examination requirements to the NRC for review and approval. The guidance should address the process for adjusting the inspection area and/or coverage interval for both welded and non-welded components as a function of the component being inspected and/or the degradation mechanism being assessed during plant-specific inspections.

With respect to inaccessible components, the MRP should:

a) Identify any components that are; (1) totally inaccessible (can't be inspected) and (2) the management of their aging effects is dependent on the inspection results from another primary, expansion, or an existing component.

b) For the components identified in "a)," identify the primary, expansion, or other existing components that are the surrogate for the inaccessible components and explain why the surrogates are the limiting components for the aging effects that need management.

c) For the totally inaccessible components or for the inaccessible portion of primary or expansion components, what are the requirements that the licensee must follow to ensure that the components do not lose their intended function as a result of flaws in the accessible components.

The response to this question should specifically propose text that would be added to the "-A" version of MRP-227 to address this issue.

Response: This RAI requests additional information on the description of inspection coverage provided in Tables 4-1 through 4-6 of MRP-227. The RAI also requests additional information on inspection coverage for Tables 4-8 and 4-9 of MRP-227; however, inspection coverage for Existing Program components listed in Tables 4-8 and 4-9 of MRP-227 is covered by the Existing Program requirements – such as the inspection coverage requirements for PWR core support structures listed in Tables 4-8 and 4-9 that are subject to ASME Code Section XI Examination Category B-N-3 visual inspections. Therefore, the response to this RAI will address only the inspection coverage requirements contained in Column 6 of Tables 4-1 through 4-6 of MRP-227.

The RAI requests very specific information to be addressed in Column 6, with the first item as "identify the location(s) where degradation is expected," with the second item as "define the appropriate inspection coverage at this location to ensure that enough of the surrounding material is inspected ....".

Because the RAI deals with the multiple facets of accessibility and coverage, the response is broken into multiple parts.

In response to the question concerning assurance that the sufficient surrounding material is inspected, the response is that all welds inspected for SCC or IASCC must be examined along with the weld heat affected zone (HAZ). This requirement is defined in

MRP-228, which states in 2.3.6.4(a): "For welds, the area of interest is generally considered the entire width of the weld and 3/4 inch of the adjacent base material on each side of the weld." This is consistent with the BWRVIP-03 "area of interest" standards for IGSCC in stainless steel welds in BWR internals.

The RAI also requests more information concerning other accessibility issues. There are several inspection strategies employed in MRP-227 to deal with accessibility and other issues that help ensure an adequate inspection sample. For the purpose of this RAI response, these are broken into the following categories:

1. Selecting 100% of accessible surfaces of a continuous structural weld,
2. Selecting 100% of accessible bolting in a bolted assembly,
3. Selecting 100% of accessible surfaces of a set of components or component items,
4. Selecting a focused examination where analysis clearly indicates which portion of the component, assembly or structure is most susceptible,
5. Selecting a sampling approach where a only portion of the component or component item is accessible without disassembly,
6. Selecting the Primary over the Expansion component where access to the Expansion component is severely limited,
7. Physical measurements for core clamping functionality.

The first situation involves those components where the examination requirement is to examine 100% of accessible surfaces of a continuous structural weld employing visual EVT-1 techniques. This requirement applies to the core barrel upper flange welds for both CE and Westinghouse designs. These welds are relatively accessible with no known significant inspection restrictions. In addition, the welds are not particularly susceptible relative to comparable BWR core shroud welds. Thus, this examination is expected to be confirmation of lack of significant degradation. However, since it is important to obtain a large sample and to ensure unanticipated accessibility restrictions are properly addressed, a minimum coverage requirement is proposed to be added to Tables 4-2 and 4-3 for these components and to be included in the '-A' version of MRP-227 (See Appendix A Point 3 of these RAI responses). For completeness, this proposed minimum requirement is also included in this RAI response:

> "A minimum of 75% of the total weld length (examined + unexamined) including coverage consistent with the Expansion criteria in Table 5-2 (or Table 5-3), must be examined from either the inner or the outer diameter for inspection credit".

As indicated above, this minimum coverage is justified by the lack of negative examination results from previous B-N-3 examinations, the anticipated low susceptibility of the core barrel welds, and an aggressive sampling approach that is considered adequate to demonstrate the continued absence of degradation.

The second situation involves the requirement to examine 100% of the accessible bolting in a bolted assembly. This applies to:

- Upper core barrel bolts and locking devices,
- Lower core barrel bolts and locking devices,
- Baffle-to-former and internal baffle-to-baffle bolt locking devices,

for the B&W designs,

- Core shroud bolts

for CE designs, and

- Baffle former bolts, and
- Baffle edge bolts

for Westinghouse designs.

For this situation, the applicable bolted assemblies were included in the irradiation modeling efforts that were a large part of the various functionality analyses. The results of these analyses provided trends and insights into the complex behavior of the structures, from which targeted examination recommendations could have been made. These results when combined with existing minimum bolting assessments and relative lack of negative experience could have justified a limited examination scope. However, it was recognized that, in order to account for uncertainties and obtain a large initial sample, a requirement of 100% of the accessible bolts was logical. This strategy serves three goals, 1) it is not limited to minimum compliance but seeks the best efforts of the examiner, 2) it diminishes the uncertainty in predictability of assembly, manufacturing uniformity, and 3) it reduces the potential for competing degradation effects to progress undetected.

Camera access to the bolt head and locking devices are, with minimal exceptions, without limitation. Thus, where visual VT-3 examination is required for bolt locking devices or general assessment of an undamaged condition, there is virtually no access limitation anticipated. The potential significant limitation on accessibility for this situation only occurs when volumetric examination (UT) is specified and a bolt head design or as-built condition limits access or effectiveness for the UT transducer. Since this is a potential severe limitation, considering the industry's limited examination experience across the entire variety of design variations, the MRP is proposing a minimum coverage requirement to be added to Tables 4-1, 4-2 and 4-3 and to be included in the '-A' version of MRP-227 to further assure that potential limitations on access and examination coverage will be adequately addressed.(See Appendix A Point 3 of these RAI responses). For completeness, this proposed minimum requirement is also included in this RAI response:

> "A minimum of 75% of the total population (examined + unexamined) including coverage consistent with the Expansion criteria in Table 5-1 (or Table 5-2, or Table 5-3) must be examined for inspection credit."

This minimum coverage requirement means that expansion of coverage when degradation is detected at that specific location is not an issue; the examination of 100% of the accessible surfaces or items will provide the necessary evidence to show whether the degradation is localized or widespread directly, without the need for coverage expansion. Also, the ability to expand from 100% of accessible surfaces or items for a Primary component to 100% of accessible surfaces or items for an Expansion component provides additional evidence about the limits of detected degradation. Throughout the process of determining the final recommendations for examination coverage, access to the Primary components has not been an industry concern and thus the coverage requirements were deemed adequate for the aforementioned reasons. However, given the potential for more severe than anticipated limitations due to the industry's limited examination experience across the entire variety of design variations, the MRP is proposing the above minimum coverage requirements to Table 4-1, 4-2 and 4-3 so that further assurance that potential limitations on access and examination coverage will be adequately addressed.

For both of the above situations, if a minimum coverage is not satisfied, the intent of MRP-227 is not met and, as discussed in RAI response 4-7, a deviation must be prepared and the staff and the MRP notified of the inability to meet the "Needed" requirement for coverage. This requirement already applies where specific recommendations are made for other Primary components and no further changes to MRP-227 need be made as discussed for the remaining situations discussed below.

The third situation involves those components where 100% of accessible surfaces of a set of components or component items is required. This applies to:

- CSS cast outlet nozzles,
- CSS vent valve top retaining ring,
- CSS vent valve bottom retaining ring,
- Lower grid assembly dowel-to-guide block welds,
- IMI guide tube spiders, and
- IMI guide tube spider-to-lower grid rib section welds

and is applicable only to B&W Primary components,

The strategy employed for these component items is to specify that the accessible surfaces of all of the components in the population be examined to the extent that it would not require disassembly. All of the components are accessible, although some portions of the visible surfaces will be obstructed. In all of these cases, the required VT-3 visual examination is looking for gross degradation, such as separation of material, broken or missing locking welds/devices, etc. Thus, no minimum coverage requirement is deemed necessary.

The fourth situation occurs where the scope of the component or the component assembly selected surface focuses on where degradation effects would be expected. The following components were selected based on insights from the materials modeling/aging analysis performed on the core barrel regions:

- CE (welded two sections) – Visual examination of shroud plate-to-former plate welds at the re-entrant core shroud mid-plane to detect cracking (IASCC).
- CE (welded full height) – Visual examination of shroud plate core-side welds at the re-entrant core shroud mid-plane to detect cracking (IASCC).
- CE (welded two sections) – Visual examination of upper and lower section flange interfaces to detect distortion.
- CE (bolted) & Westinghouse – Assembly level visual examination to detect distortion.

For these components, the areas selected consist of the locations where the highest predicted void swelling would manifest itself as gaps and/or displacements at joint interface locations or where the highest irradiation/stress combination would most likely produce cracking due to IASCC. Thus, no minimum coverage requirement is deemed necessary.

The fifth situation occurs where the recommendations involve a sampling strategy employed specifically due to access limitations. One of the basic assumptions is that disassembly of reactor internals would be avoided unless warranted. This sampling examination strategy has been specified for:

- B&W – visual examination around baffle plate bolts holes ("100% of the accessible surface within 1 inch around each flow and bolt hole"). This recommendation provides for examination of an adequate amount of surface area with the bolt holes selected as the most likely point of initiation due to the bolt hole and locking weld acting as a stress riser.
- CE – visual examination of deep beam-to-beam welds axially from top surface to four inches below. This results in a sample of the top four inches of 100% of the deep beam intersections which equates to approximately 10-15% of the available weld length. However, the sample size selected provides reasonable confidence in detecting the extent of potential degradation effects as it encompasses the most highly irradiated portion of the component.
- CE – visual examination of outer peripheral guide tubes of the control element assembly instrument guide tubes. This recommended scope can be accomplished without by unnecessary disassembly with an adequate sample size selected to provide a reasonable level of confidence in detecting the extent of potential

degradation effects. The sample size is expected to include roughly 15-20% of the population.

- Westinghouse – visual examination of all CRGT guide cards within a 20% CRGT sample. This recommendation avoids unnecessary disassembly with an adequate sample size selected to provide reasonable confidence in detecting the extent of potential degradation effects.
- Westinghouse – visual examination of the outer CRGT lower flange welds. This recommendation avoids unnecessary disassembly with an adequate sample size selected to provide a reasonable level of confidence in detecting the extent of potential degradation effects. The sample size is expected to include roughly 15-20% of the population. All of the guide tube weld locations are equal in their relative susceptibility.

The sixth situation involves choosing as Primary the accessible component over an inaccessible Expansion component. These three cases always elevate a component of equal or greater susceptibility that is accessible over the inaccessible Expansion components. In no instance is a more susceptible component placed in the Expansion pool due to inaccessibility. This, coupled with the conservative threshold for requiring the evaluation of the inaccessible Expansion components, provides reasonable assurance that the appropriate evaluation and/or corrective actions will be taken well before significant degradation occurs.

The seventh situation involves physical measurements to demonstrate acceptable core clamping functionality for the B&W and the Westinghouse designs. These components are accessible to perform these measurements.

In addition, plant-specific fatigue evaluations are to be employed for 3 CE components that may have some access limitations. The results of this evaluation will determine what, if any examinations are recommended. See RAI 4-16 response.

Rules concerning the acceptance and disposition of flaws in a component where limited examination coverage is or can be obtained are contained in WCAP-17096-NP. This document has been submitted for review and approval.

Finally it must be mentioned that one of the Primary component items has recently been determined to be inaccessible. As mentioned in our October 14, 2010, meeting with the staff, AREVA has been working closely with the B&W owners in preparation for their implementation of MRP-227. This effort has included record searches and more meticulous accessibility studies. The result has been that the CSS vent valve disc shaft (or hinge pins) cannot be seen without disassembly of the valve. The portions believed accessible are in fact covered by other journal bushing rings. This will result in a change to Table 4-1 of MRP-227 to evaluate the condition or replacement. As discussed in the B&W supporting documents, these disc shafts/hinge pins are subject to loss of fracture toughness due to thermal embrittlement. As indicated in the Table 4-1 footnotes, these valves are exercised each outage to ensure they are functional, i.e. that they will lift within the design limits. Additionally, failure during operation that results in valve displacement would be detected by operators as abnormal by-pass flow. These factors offset the concern with inaccessibility, but the MRP recognizes that a formal evaluation -- either generic for the B&W fleet or on a plant-specific basis -- should be completed or a replacement strategy recommended. Thus, in the interim, Table 4-1 will be changed to indicate that the CSS vent valve disc shafts (or hinge pins) are inaccessible and that an evaluation or a replacement campaign is required (See Appendix A Point 4 Change a of these RAI responses).

RAI 4-9: A number of components are identified as being covered by existing programs. However, there is no summary of existing RVI programs provided in MRP-227 or supporting documentation. Add a summary of existing programs being credited to MRP-227. If an existing program is consistent with a program definition given in the staff's Generic Aging Lessons Learned (GALL) report, it is sufficient to simply identify the related GALL program definition. For existing programs lacking such a convenient reference, a summary should be provided which describes the following program requirements for an acceptable existing program: (a) scope (i.e., components inspected/monitored), (b) the applicable inspection, monitoring, or testing requirements and acceptance criteria, (c) the periodicity of the program, and (d) any other relevant requirements. This summary should identify the degradation mechanism(s) that are intended to be monitored or mitigated by each existing program and provide justification that each program is sufficient to monitor and/or mitigate all the expected degradation mechanisms identified in MRP-227 for the applicable component(s).

The response to this question should specifically propose text that would be added to the "-A" version of MRP-227 to address this issue.

Response: First, as stated in MRP-227 (Section 4.4), no existing generic industry programs were considered sufficient for monitoring the aging effects addressed by these guidelines for B&W plants. Therefore, no components for B&W plants were placed into the Existing Programs group.

Second, Tables 4-8 and 4-9 of MRP-227 identify and provide a reference to all credited Existing Program components for Combustion Engineering and Westinghouse plants, respectively. All of the credited Existing Programs in both tables are covered by the visual (VT-3) examination requirements of Examination Category B-N-3 of the ASME Code Section XI, except for the flux thimble tubes in Westinghouse plants. For those components covered by ASME Section XI, GALL report Section XI.M1 provides an adequate summary of program requirements, and the ASME Section XI inspections specified in MRP-227 are consistent with GALL XI.M1 program requirements. The

Table 4-9 reference for flux thimble tubes in Westinghouse plants is IEB 88-09, which is based on individual licensee commitments. The MRP-227 recommendations are consistent with the requirements of the GALL XI.M37 Flux Thimble Tube Inspection program, and site-specific commitments are included within this program.

Not included in Table 4-8 are the references for Existing Programs for ICI thimble tubes and thermal shield positioning pins in CE plants. Again, the guidance for these components is limited to plant-specific recommendations that preclude the preparation of a generic summary of program elements for all plants. This is consistent with line item IV.B3.RP-357 in GALL Rev. 2 which calls for a plant specific aging management program for ICI thimble tubes. Plant owners will review their plant-specific commitments when developing individual plant aging management programs for these components. A similar situation applies to guide tube support pins (split pins) in Westinghouse plants. Plant-specific recommendations involve owner/operator replacement decisions and not an inspection program. As a result, the decision was made to exclude these components from Table 4-8 and 4-9 for the CE and Westinghouse designs respectively.

The staff may wish to continue to include the plant-specific disposition of the ICI thimble tubes, and thermal shield positioning pins for CE plants and the guide tube support pins as plant-specific actions for license renewal submittals.

RAI 4-10: MRP-227 guidance is used to develop component-level aging management programs and inspection requirements. Further, the development of these programs and requirements has not considered the effects of transitory design basis events (DBE) on the performance of degraded components or structures. However, as indicated in the response to RAI 2-16, the current industry expectation is that "...when age-related degradation effects are detected during the examinations specified in MRP-227, the suitability of the degraded component for continued service will necessarily take into consideration the full range of design basis event (DBE) effects." Therefore, staff believes that guidance and requirements should be provided to ensure that licensees perform consistent plant-specific evaluations of the effects of degraded components under both normal and transitory DBEs (i.e., normal, upset, emergency, and faulted loading conditions). These evaluations should provide reasonable assurance that the systems associated with the degraded components will maintain required design margins and that inspection, repair, and replacement requirements are both adequate and timely. The guidance and requirements should, in part:

a) Identify the number or percentage of related primary or expansion components that should be inspected and the allowable number of degraded or non-functioning components for each system to ensure acceptable performance under DBEs. Discuss the appropriateness of developing generic versus plant-specific inspection requirements for each system. Alternatively, the guidance should describe how the plant-specific analysis should be performed to determine the inspection sample and allowable number of degraded components for each system, and

b) Describe the additional inspection requirements that should be triggered if degraded components are found as part of the primary inspection.

The consideration of item b) should provide guidance for increasing the sample size to inspect other similar components within the system that are subject to the observed degradation mechanism. It should either provide guidance for expanding the inspection to components in other systems that are subject to the same degradation mechanism or justify the adequacy of existing expansion criteria in MRP-227. As an example, this guidance should specify the percentage of baffle-to-former bolts that should be inspected and the percentage that may be degraded before system performance under design basis loading conditions is affected. If degraded baffle-to-former bolts are found, this guidance should next specify the additional baffle-to-former bolts that should be inspected and, for instance, the number of expansion baffle-to-baffle or core barrel-to-former bolts that should be inspected.

Response: This RAI expresses the staff's preference that "guidance and requirements should be provided to ensure that licensees perform consistent plant-specific evaluations of the effects of degraded components under both normal and transitory DBEs (i.e., normal, upset, emergency, and faulted loading conditions)." The MRP recognized the industry's need to have consistent methodologies for evaluating the results of inspection findings from implementation of MRP-227. As a result MRP requested that the PWROG work with the industry to provide guidance for component/component assembly acceptance and evaluation. WCAP-17096, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," Revision 2, December 2009, was developed and submitted to NRC for review with the intent that it provide "consistent, industry-wide analytical methodologies and data requirements for developing:

1. **Acceptance Criteria** for the Primary and Expansion Components identified in the Materials Reliability Program (MRP) Reactor Internals Inspection and Evaluation (I&E) Guidelines (MRP-227 Rev. 0)

2. **Evaluation Procedures** for utilities to assess potential safety and functional impacts of degradation in components with observed relevant conditions."

MRP-227 does provide guidance on expanding the inspection sample size when degradation is detected in excess of the Expansion criteria in Section 5. That increased sample size is the main purpose behind the sampling strategy represented by the Primary components and the potential enlargement to include Expansion components. Both the Primary and Expansion component inspection coverage in MRP-227 specifies the percentage (100%) of accessible bolts that should be inspected.

WCAP-17096 has been submitted to the staff for review and evaluation, and detailed information requests regarding evaluation of examination results can be addressed during its review. In order to ensure that consistent methodologies are employed by utilities in the evaluation of degradation the MRP will add a 'Needed' requirement to the approved version of MRP-227 (see Appendix A to these RAI responses).

It is the MRP's expectation that flaw evaluations performed in accordance with NRC-approved methodologies do not require transmittal to or approval by the NRC. However, under no circumstances would any requirements to report flaws or flaws evaluations contained in ASME Section XI be superseded. Flaw evaluations that do not meet NRC-approved guidance (e.g. assumptions, methods, etc.) are to be submitted for approval.

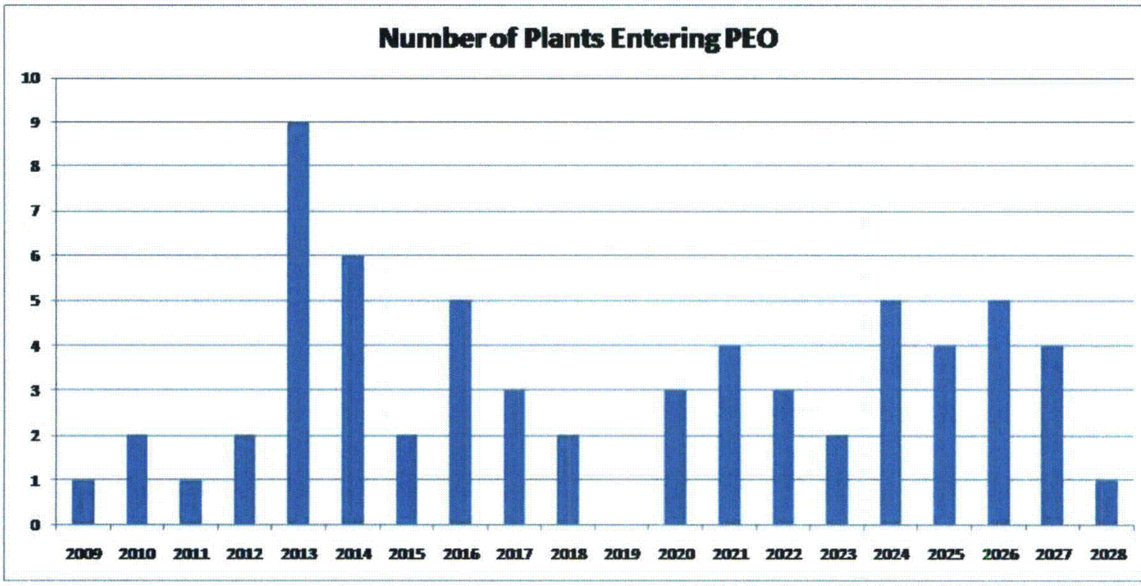
MRP-227 does provide guidance on expanding the inspection sample size when degradation is detected in excess of the Expansion criteria in Section 5. That increased sample size is the main purpose behind the sampling strategy represented by the Primary components and the potential enlargement to include Expansion components. Both the Primary and Expansion component inspection coverage in MRP-227 specifies the percentage (100%) of accessible bolts that should be inspected.

RAI 4-11: RAI 2-1 asked for justification of the inspection periodicity recommended in MRP-227 for reactor vessel internal components given that there is little operating experience for basing inspection periodicity and that analysis to evaluate the evolution of degradation in these components has a large degree of uncertainty. The response to that question primarily justifies the adequacy of the recommended inspection intervals based on the functionality analysis which predicts that degradation will gradually worsen over time and will not suddenly progress. However, the inspection periodicity is not based, as is typically the case, on an evaluation of the maximum level of degradation that is acceptable for components to fulfill their intended design requirements, and the predicted time to reach this level of degradation based on the extent of degradation found during the inspection and evaluation of the rate of degradation with continuing operation. Therefore, the staff requests additional justification for recommended inspection intervals. This justification should address why the current MRP-227 approach is appropriate for determining inspection periodicity and that determining periodicity based on a component level evaluation to ensure that the required component design margins are retained between inspections is not required.

Alternatively, provide information about the plans of licensees to perform the initial primary inspections required by MRP-227. Staff understands that some licensees are planning to inspect all required primary components during the first refueling outage after entering into the license renewal period. Staff therefore seeks to determine if this approach is being adopted by other licensees that have or will shortly enter the license renewal period. Clarify if this approach is either recommended or required within MRP-227.

> **Response:** The response to RAI 2-1 (Set #2) described the results of the component degradations in the functionality analyses as very gradual, with no sudden increases during either the 30 years of simulated high-leakage core loading or the subsequent 30 years of low-leakage core loading. From this work, it was determined that: (1) inspections immediately prior to or immediately following the beginning of the extended period of operation would have similar likelihoods of detecting and characterizing any degradation present in the reactor internals Primary components, and (2) no basis for departure from the standard ten-year inspection intervals to more frequent inspection intervals was warranted. The latter determination was supported by both operating experience and the results of ASME Section XI Examination Category B-N-3 inspections conducted during the first 40 years of operation. In addition, since MRP-227 is a living document, which will be updated to reflect both positive and potentially negative information from inspection results obtained by a series of plants entering the period of extended operation over the next several years, the currently-specified inspection periodicity will be reviewed and altered, as needed, to incorporate that information. As currently anticipated, some five or six plants will be conducting all or portions of the inspections specified in MRP-227 in the next four years. Additionally, as can be seen graphically below, the US PWR fleet will have units entering their period of extended operation over intervals sufficient to ensure examination results are regularly being collected.

Demand for MRP-227 SER
PWRs Entering Period of Extended Operation



Even more to the point, the inspection results could result in the detection and sizing of defects that might require engineering evaluation for disposition. In such cases, the engineering evaluations will provide clearer evidence than is now available on the level of degradation that is acceptable, and subsequent inspections will provide information on the rate of increase in that degradation. This type of information will provide additional grounds for sustaining or modifying the current inspection periodicity. Considering the operating history of PWR internals in the United States, the results of ASME Code prescribed examinations on internals over the first 40 years of operation, and the conservative results from the functionality analyses, the current inspection periodicities are considered consistent with the intent of the 10CFR54 requirements. Further, on the same basis, they are considered adequate to identify potential degradation of the non-pressure boundary components associated with the reactor internals with the potential for either increasing or decreasing that periodicity as warranted by initial examination results.

Several plants have provided their plans to perform the initial primary inspections required by MRP-227 to the NRC. Some licensees are planning to inspect all required primary components during the first refueling outage after entering into the license renewal period. This approach is not a direct recommendation within MRP-227-Rev. 0, as the inspection Tables (Tables 4) in MRP-227 provide some flexibility so that a PWR

Unit does not necessarily have to inspect all of its MRP-227 Primary components during the same outage.

The following inspection plans were provided by owners depicting their current plans to perform the initial primary inspections required by MRP-227 to the MRP are shown below.

A Westinghouse 3-loop has provided the following current plan for information.

Inspection	2010	2011	2012	2013	2014
Baffle bolts UT	(1) W 3 loop	(1) W 3 loop			
B&W Core Barrel bolts UT					
Primary EVT-1			(2) W 3 loop		
Primary VT-3 or measurements			(2) W 3 loop		
Primary VT-3 requiring CB removed				(1) W 3 loop	(1) W 3 loop

3 B&W Units have provided the following current plan for information. These units currently plan to perform the "Primary Components" inspection from MRP-227, Rev. 0 Table 4-1 during the fall 2012, fall 2013 and spring 2014 respectively, with the exception of two examinations already performed and possible changes from a deviation in progress. The two examinations that have been performed are the onetime physical measurement of the interference fit between the plenum cover weldment rib pads and RV flange, and the UT of the upper core barrel bolts. The timing of the examinations is given in the table below:

Physical Measurement		
Unit-1	Fall 2006	No change From Base Line during initial installation
Unit-2	Fall 2008	No change From Base Line during initial installation
Unit-3	Fall 2007	No change From Base Line during initial installation
UT of Upper Core Barrel Bolts		
Unit-1	Spring 2008	100% coverage, no rejections
Unit-2	Fall 2008	100% coverage, no rejections
Unit-3	Fall 2007	100% coverage, Two bolts rejected, lack of back wall, same as previous inspections

The deviation moves two "Primary" components (Vent Valve Disks and ONS-3 Cast Outlet Nozzles) to "No Additional Measures" due to ferrite content being below the screening criteria, and one "Expansion" component (CRGT Spacer Castings) being moved to "Primary" as a result of the other two castings being moved to "No Additional Measures".

RAI 4-12: RAI-21 (Set #2) asked about the need to develop more definitive acceptance criteria for inspections to ensure uniform interpretation and implementation from plant to plant. The industry response indicated that more definitive criteria in MRP-227 is not needed because the inspectors will receive component-specific training and that any observable degradation will require further disposition through the corrective action process. However, staff remains concerned that this approach is not sufficient given the variability associated with inspection conditions and interpretation of inspection results. Therefore, staff requests that more definitive inspection acceptance criteria be developed for the VT-1, EVT-1, UT, and VT-3 inspections for each of the primary and expansion components. These criteria should be a function of the required accuracy and precision of the particular technique and also the application of this technique to each particular component (i.e., accessibility limitations, expected degradation location, expected degradation type).

In the absence of adequate guidance, the staff will consider the need to implement limitations and conditions on the use of MRP-227 which would address plant-specific action items necessary to address this issue for each facility.

> **Response:** In the response to RAI 2-21 (Set#2) the MRP described the inspection acceptance criteria contained in Section 5 of MRP-227 for VT-3, VT-1, EVT-1, and UT examinations, which are relatively simple; any detected relevant condition must be reported and placed in the plant corrective action program. Such simple inspection acceptance criteria will be interpreted and implemented consistently at different plants. MRP-228 has been developed and issued to ensure exactly that.
>
> The MRP agrees that establishing more definitive acceptance criteria leads to more uniform and consistent plant-to-plant implementation especially for signal-related examinations.
>
> The MRP believes that adequate direction has been provided for the ultrasonic examination of bolting by requiring that the technical justification process from MRP-228 (based on ASME Section V Article 14) be followed. A substantial discussion of this process is included in Section 2 of MRP-228. This discussion addresses: (1) whether a formal Technical Justification is required or not (some examination methods are sufficiently standardized and codified that no addition technical justification is needed); (2) the required accuracy and precision that must be demonstrated by the Technical Justification, including the potential for mock-ups that simulate the range of geometries and types of degradation needed; and (3) the personnel training that is planned to ensure some degree of uniformity in implementation and interpretation.
>
> Definitive (numerical) acceptance criteria is provided for the physical measurement specified for B&W plants (height differential from top of plenum ribs pads to RPV

seating surface) due to uniformity of design characteristics. The physical measurement for Westinghouse plants is the spring height for those plants with Type 304 stainless steel hold down springs. While there is similarity in the basic design for the Westinghouse plants, the numerical acceptance criteria is a plant-specific input due to the variations in designs loads and is also plant-specific as-built dimensions. There are no direct physical measurements specified as Primary or Expansion for CE components.

The MRP believes that uniformity of inspection implementation can be accomplished without technical justifications for visual examinations based on the uniformity of existing industry requirements and Section XI of the ASME Boiler and Pressure Vessel Code (Section XI) examination experience. The MRP requirements for visual examinations have incorporated Section XI and BWR Vessel and Internals Project (BWRVIP) EVT-1 requirements with augmented guidance based on expected or potential degradation. The visual examination rules of Section XI are in IWA-2210 (2010 Edition). These rules are implemented routinely across the entire fleet virtually every outage with oversight by NRC and Authorized Nuclear Inservice Inspectors. Consistent implementation has been established. The BWRVIP developed the EVT-1 examination by utilizing the Section XI visual rules and augmented them where needed based on the type flaws to be expected. The essential elements of the BWRVIP developed EVT-1 will be used by MRP.

Over the last 16 years enhancements have been made to the EVT-1 examinations based on lessons learned that were shared by all member utilities via the BWRVIP Inspection Focus Group. Additional insights were gained from INPO review visits where peers identified best practices. These lessons are reflected in current BWRVIP EVT-1 guidance and have been utilized by the MRP. Also, there is current research and testing ongoing regarding visual examination methods. MRP is integrated into the research and will gain from insights that result. MRP will also share lessons learned via its Inspection ITG and through INPO best practices. Where appropriate, the visual examination guidance will be revised to reflect industry experience. Finally, the current, known inspection vendors have performed visual examinations per Section XI and the EVT-1 requirements for many years further ensuring consistent implementation of visual examinations when employed in implementation of MRP-227.

Providing explicit criteria for relevant conditions from visual examinations has the disadvantage of potentially biasing examiners and owners so they inadvertently disregard unanticipated conditions. Considering the scarcity negative OE for reactor internals to date and the fact that many of the Primary visual examinations will be the first instances where a component is inspected, it is important to get a good baseline. To do this and to anticipate the potential for these initial examinations to find the "unexpected", MRP required that <u>any</u> relevant condition be reported. Explicit relevant conditions are specified when evaluating the need to expand from the Primary to Expansion components. Relevant conditions not specified in the Primary to Expansion linkage would not require expansion to comply with MRP-227; however all relevant conditions require reporting and disposition via a plant's corrective action program. Corrective action programs typically require that consideration of extent of condition be provided as part of dispositions. Aligning MRP-227examinations with well proven, well known and used ASME methods of examination and terminology is anticipated to provide more information reporting and subsequent industry scrutiny.

RAI 4-13: Provide a description of how international and US operating experience is (or is planned to be) documented, tracked, and updated so that it will support continued refinement of MRP-227 guidance and inform plant-specific aging management programs.

The staff believes that it would be advisable for documentation regarding US and international operating experience related to degradation in RVI components be compiled in a single document that could be used to support the process of updating MRP-227.

Response: The MRP proposes to use a new Appendix A of MRP-227, which currently contains a summary of US and relevant international operating experience related to degradation of RVI components to date (this was provided to the NRC by MRP Letter 2001-091 dated December 2, 2009; see Appendix A to these RAI responses). This summary was compiled during the preparation of the supporting documentation for MRP-227, and was generated with the intent to combine and place this information in a single location that would be periodically updated with additional operating experience as it becomes available. Section 7.6 of MRP-227 identifies a "Good Practice" recommendation that will help to assure the collection of this operating experience. The MRP recognizes the importance of documenting the operating experience to support the maturation of processes, procedures and requirements in the family of documents centered around MRP-227 that will implement the inspection and evaluation requirements. Thus the MRP proposes that the "Good Practice" recommendation be modified in Section 7.6 to elevate to the "Needed" requirement to provide inspection results. That recommendation would be changed to state that (see Appendix A to these RAI responses):

> "*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 are examined.*"

This recommendation is similar to those provided under other NEI 03-08 Issue Programs (IP), e.g. periodic BWRVIP Inspection Summaries, MRP Inspection Survey. While a final format and method of sharing has not been determined the information obtained from these summary reports will be available to update the operating experience record on a periodic basis. The current format for summarizing and updating that operating experience may continue to be the current format contained in Appendix A of MRP-227, or the format may change to parallel one of the other IPs, or that format may change to adapt to circumstances. It is also the expectation that these summary reports will be provided in similar fashion to the staff either as part of a stand-alone document or combined with other reporting of inspection results (i.e., MRP-219).

Because foreign utilities are not as obligated as part of the NEI 03-08 initiative, obtaining international experience is not as certain. Nevertheless, foreign utilities actively participate in the MRP and the PWROG-MSC and share inspection experience in meeting forums, as well as cooperative efforts to benchmark each others approaches to the managing the aging of reactor internals.

RAI 4-14: Verify that neither MRP-210, "Materials Reliability Program: Fracture Toughness Evaluation of Highly Irradiated PWR Stainless Steel Internal Components," nor Section 6 of MRP-227 will be used to disposition (i.e., determine need to repair, need to replace, or inspection periodicity) degraded components identified during RVI inspections and that this guidance will instead be provided by WCAP-17096, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," Revision 2, December 2009. If this is not the case, please provide a description of the relationship between MRP-210, Section 6 of MRP-227, and WCAP-17096 and identify the aspects of the disposition analysis that will be governed by each document.

Response: Demonstration of the acceptability for continued service of PWR internals components is required when a relevant condition exceeding the examination acceptance criteria of Section 5 of MRP-227 is detected.

MRP-210, "Materials Reliability Program: Fracture Toughness Evaluation of Highly Irradiated PWR Stainless Steel Internal Components," is one of the key and informative reference documents that was used in the preparation of MRP-227, and Section 6 of MRP-227 is considered to be valuable information that could be useful to PWR licensees. However, neither MRP-210 nor Section 6 of MRP-227 contain any "Mandatory" or "Needed" requirements for engineering evaluations of degraded components identified during the examinations and inspections specified in Table 4-1 through 4-9 of MRP-227. MRP-210 was intended to demonstrate the relative flaw tolerance for potential flaws in several typical reactor internals components including irradiated material considerations but was not intended to establish a particular component's critical flaw size nor establish generic requirements.

On the other hand, WCAP-17096, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," Revision 2, December 2009, provides generic methodologies tailored more specifically to the component or assembly and the aging effect(s) that can be used by licensees to generate engineering evaluation acceptance criteria. This is the document that will be used as the framework to develop those more specific evaluations either generically where similarities support generic efforts or for plant-specific applications.

RAI 4-15: There have been several previous RAIs related to cast austenitic stainless steel (CASS) materials, but several questions/issues remain.

a. The industry is currently supporting using a minimum irradiation embrittlement (IE) threshold of 1 displacement per atom (dpa) to determine susceptibility of CASS components to IE, yet available data seems to support a threshold of 0.3 dpa or less as there is little data between 0.05 and 1 dpa and current data indicates some toughness decrease between 0.3 and 1 dpa. Would a reduction in the screening threshold from 1 to 0.3 or 0.05 dpa result in additional components screened in for IE? If so, identify the CASS components that would likely be screened in for IE susceptibility due to these lower screening thresholds. Finally, many CASS components are in the A inspection category. Please provide the basis/justification for placing these components in the A category.

b. The fracture toughness of CASS can degrade significantly due to thermal embrittlement (TE) and the toughness of both CASS and other stainless steel materials can decrease significantly as neutron fluence increases. When the dose exceeds 5 dpa, available data indicates that fracture toughness can be extremely degraded in many materials. The staff's concern is that the fracture toughness in CASS components may get so low due to TE and/or IE and in other components due to IE that preexisting fabrication or service-induced flaws that are smaller than the inspection resolution may challenge component integrity under normal loading or under design basis events. Additional guidance to licensees may be needed either in MRP-227 or WCAP-17096, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," Revision 2, December 2009, to address this situation. Please describe how existing or planned guidance addresses this issue. Otherwise justify why such guidance is not needed.

Response: The MRP has recently completed an assessment of thermal aging and irradiation embrittlement for the CASS materials in PWR internals (MRP-276). In response to item **RAI 4-15a**, the following information has been excerpted from this report:

There are three CASS items in the B&W design PWR internals that are expected to exceed a fluence of $1x10^{17}$ n/cm^2, E > 1.0 MeV at the end of a 60-year lifetime. These three items are:

- IMI guide tube assembly spiders
- CRGT assembly spacer castings
- Plenum cylinder reinforcement castings (DB only)

The IMI guide tube assembly spiders are classified as a primary item in MRP-227. A visual (VT-3) examination of the IMI guide tube spiders is to be performed. The IMI guide tube spiders are being examined to detect spider arms that do not align with the lower fuel assembly support pad center bolt. The recommended methodology for acceptance given in WCAP-17096 is to perform an analysis to show that one or more missing spider arms or a completely missing spider will not result in loss of function of the IMI guide tube. Thus, no fracture mechanics evaluations are needed. Since there are 52 IMI guide tubes in each B&W unit, a redundancy argument may also be adequate.

The CRGT assembly spacer castings are classified as an expansion item in MRP-227. If necessary, a visual (VT-3) examination of the CRGT spacer castings is to be performed. The spacer castings have limited accessibility from the top or bottom of the CRGT through a center free-path (once the plenum assembly is removed from the vessel). Examination at the quarter points where the threaded connections are present is possible. These lanes are not blocked by the rod guide tubes. The examination would look for cracking of the spacer surface or evidence that the spacer is not approximately centered. The threaded fasteners are welded to the OD of the pipe column so it is possible that a degraded threaded location would not be detected. Since there are 69 CRDMs in each B&W unit, the recommended methodology for acceptance given in WCAP-17096 is to perform a reactivity analysis to determine the number of CRDMs that are required for shut down of the reactor. Thus, no fracture mechanics evaluations are needed.

There are two plenum cylinder reinforcement castings (at DB only), which have not been included in any MRP or PWROG evaluations as of the preparation of this document. As noted in Table 4-1 of MRP-276, only the lower edge of these castings are expected to exceed a fluence level of $1x10^{17}$ n/cm^2, E > 1.0 MeV at the end of a 60-year lifetime. Assuming the evaluations and conclusions would be the same as the wrought reinforcement plates at the other B&W units, this item would be classified as "No Additional Measures." It is also possible that this item could be dispositioned by reviewing the materials records and determining the ferrite content to be below the

MRP-175 screening criteria, which would also classify it as "No Additional Measures."

Therefore, it is concluded that the CASS items in the B&W design PWR internals are redundant and/or potentially able to be analyzed for functionality in the anticipated degraded conditions. Replacement of the degraded item or component is also a potential option. Thus, no fracture toughness properties would be required for fracture mechanics analyses.

MRP-191 indicated there are only four components fabricated from CASS in the CE design PWR internals:

- Core support columns
- CEA shrouds
- CEA shroud bases
- Modified CEA shroud extension shaft guides

The aging management strategy employed for thermal embrittlement in the CE reactor internals requires inspection for cracking at appropriate locations. As the thermal embrittlement susceptibility of a CASS component is determined by the ferrite content and service temperature, it is not possible to use cracking as a leading indicator of this degradation mechanism. Therefore the classification of the CASS components as primary or expansion is determined by the relative susceptibility to cracking. In the case of the CE design, there are no CASS components that are ranked as a primary component for any of the cracking mechanisms (fatigue, SCC or IASCC). The core support columns were dispositioned as an Expansion inspection component in MRP-227 and evaluation of IE and TE must be considered in the evaluation of cracking in this component.

The three remaining internals components (CEA shrouds, CEA shroud bases, and modified CEA shroud extension shaft guides) were not originally screened in based on the MRP-175 threshold fluence. These components would require additional evaluation if the lower threshold for thermal embrittlement were employed.

The CEA shroud bases accumulate enough neutron fluence to be categorized as part of Region 2 per MRP-191. Components in fluence Region 2 are expected to receive between $1x10^{20}$ and $7x10^{20}$ n/cm^2, E > 1.0 MeV (0.15 to 1.1 dpa) by the end of the 60-year license period. Thus, the CEA shroud bases are expected to exceed the NRC screening level for synergistic embrittlement. However, CASS versions of the bases are present in only one plant. All other CE plants with this component have bases fabricated from wrought Type 304 stainless steel. Consideration of the shroud bases in a generic manner is not appropriate, since CASS shroud bases are present in only one plant.

The modified CEA shroud extension shaft guides are located at the top of the CEA shroud assembly approximately in line with the mating surface between the reactor vessel and the reactor vessel head. No specific fluence or stress analyses have been completed for this component, but in this region of the reactor, a very low accumulated fluence (probably lower than the 0.05 dpa screening value) would be expected. Also, this component does not serve as a core support structure, and the loads are expected to be quite low. The knowledge gap for this component is the lack of analyses that address effects of fluence or stress. However, results of any analyses are expected to show low fluence and low stress. MRP-191 classified this component as a Category A component because of the low likelihood for failure.

The CEA shrouds constitute part of a core support structure in the CE design and may receive enough fluence to exceed the 0.05 dpa level. An initial survey of the CE plants indicated that a significant portion of the CE fleet has CEA shrouds fabricated from cast stainless steel. The FMECA panel moved this component to Category A based on a low likelihood of failure. It should also be noted that the CEA shrouds are generally centrifugal castings containing less than the MRP-175 thermal embrittlement screening criterion of 20% delta ferrite. This, in addition to the expected low likelihood of failure, supports the placement of the CEA shrouds in Category A.

There are relatively few CASS components in the Westinghouse design PWR internals. CASS was primarily used in non-structural or redundant components. Therefore, there are relatively few requirements for fracture toughness determinations. Six of the eight CASS components were already identified in the screening process for potential irradiation embrittlement concerns due to relatively high peak neutron fluences. However, there were no requirements for flaw tolerance analyses in the evaluation procedures for these components; therefore, there were no requirements for fracture toughness data. Thus, there are no data gaps identified for these six screened-in components.

There are two remaining CASS components that were not identified for irradiation embrittlement:

- Intermediate flange in the control rod guide tube assemblies
- Lower support castings

The intermediate flanges on the control rod guide tube assemblies are not expected to exceed the 0.05 dpa screening limit because these flanges are well removed from the core. Cracking in the intermediate flanges of the control rod guide tube assembly is less probable than cracking in the lower flanges, where both fluences and bending stresses are expected to be higher. At most, the intermediate flange would be an expansion item for the lower flange. Should a crack be observed in an intermediate flange in this component, a full flaw tolerance analysis would not likely be required. Rather than performing crack growth predictions, a relatively small component like the relevant portions of the intermediate flanges is typically considered failed once a crack is detected. Therefore, there is no apparent need for fracture toughness data for the intermediate flange material.

A small fraction of Westinghouse plants have lower support castings rather than lower support forgings. Although the lower support casting is well-removed from the core, there is a remote possibility that portions of this component may experience fluences greater than 0.05 dpa. In the unlikely event that cracking is observed on the surface of the lower support casting, a flaw tolerance analysis might be undertaken. Since it is a large component, it may be possible to show by analysis that the stresses at the location of the crack are too low to drive crack propagation. If a flaw tolerance analysis is conducted, an estimate of the fracture toughness in the lower support casting would be required. It may be possible to assume a lower bound toughness equivalent to highly irradiated austenitic steel and demonstrate that significant margin against failure remains.

There is a potential gap for fracture toughness data to evaluate flaws in CASS lower support castings. However, there is a reasonable potential for demonstrating structural integrity with suitably conservative assumptions. There are no other data gaps for CASS components for the Westinghouse design.

In response to item **RAI 4-15b**:

There are no CASS items in the B&W design internals that are anticipated to exceed 5 dpa (3.3×10^{21} n/cm^2, E > 1.0 MeV) at the end of license renewal.

Due to the conservative nature of the screening process, peak fluences in the Westinghouse BMI cruciforms and lower support column bodies as well as the CE core support column bodies were assumed to be greater than 3.3×10^{21} n/cm^2, E> 1.0MeV). All three of these components were placed in Category B and embrittlement effects were considered in the aging management recommendations. The Westinghouse BMI

Cruciforms were eventually placed in the "no additional measures" category because there were no identified cracking mechanisms and the cruciform has minimal structural significance. The support columns in both the Westinghouse and CE designs were identified as Expansion components to be inspected for potential cracking. Although it is recognized that these columns are potentially subject to both irradiation and thermal embrittlement, WCAP-17096 anticipates that the acceptance criteria for these columns will not take structural credit for any cracked support column. The proposed WCAP-17096 methodology requires an analysis to demonstrate that the remaining unflawed columns will provide the required structural support. This approach is similar to the minimum acceptable bolting patterns applied to core baffle and core shroud bolts. Because there is no flaw tolerance requirement in the proposed evaluation methodology, there is no need to estimate the fracture toughness of the embrittled component. The margins applied to the WCAP-17096 evaluation methodology are intended to account for undetected and newly initiated flaws.

RAI 4-16: MRP-227 identifies several components that require plant-specific aging analysis (e.g., fatigue analyses) to determine the appropriate inspection category. However, MRP-227 does not discuss or reference approved methods or acceptance criteria for conducting such analyses. Discuss why guidance is not necessary to ensure consistent application and interpretation of plant-specific aging analyses. Alternatively, if the industry plans to provide such guidance, discuss the plans, approach, and schedule for developing this guidance. This discussion should address how environmental effects should be treated in these analyses.

Response: Three Primary components for Combustion Engineering (CE) plants in Table 4-2, which were identified for potential fatigue related degradation require inspection when adequate fatigue life cannot be demonstrated by TLAA. This reference recognizes the potential for a CE plant to have a current licensing basis that includes core support structure fatigue design calculations. In such a case, the requirements for aging management are already defined by § 54.21(c) of 10 CFR Part 54. Any subsequent inspection requirements for these components TLAA would arise from the TLAA process.

The entry in the Examination Coverage column of Table 4-2 for these three components refers to a plant-specific fatigue analysis. In reviewing this RAI, it has become apparent that the wording of this entry could possibly be misconstrued. The intent of this entry was to require the utility to identify the potential location and extent of fatigue cracking. There was no intent to require a utility that was not subject to the TLAA requirements to complete a full fatigue evaluation prior to the inspection. Therefore we would propose to replace the words "plant-specific fatigue analysis" with the words "evaluation to determine the potential location and extent of fatigue cracking."

There is already adequate guidance on fatigue analysis available from both regulatory sources, such as Chapter X of NUREG-1801, or from industry sources, such as NEI 95-10. While this guidance normally addresses pressure boundary components, the same methodology and procedures, is readily applied to core support structures with a fatigue design basis. We do not believe that there is any need to duplicate this guidance in MRP-227. This same guidance can be used by plant owners that have yet to apply for license renewal. Most, if not all, of these plants are more recent vintage plants with well-defined ASME Class 1 fatigue design bases.

RAI 4-17: MRP-227 Tables 3-1, 3-2, and 3-3 identify component/aging mechanism combinations (where the combination is identified as either "primary," "expansion," or "covered by existing programs" in the tables) that are not identified in Tables 4-1 through 4-6, 4-8 and 4-9. Tables 3-2 and 3-3 also identify components (e.g., In-Core Instrumentation Thimble Tubes in CE internals and control rod guide tube support pins in Westinghouse Internals) that are not identified in Tables 4-1 through 4-6, 4-8 and 4-9 at all. Identify the component/aging mechanism combination in Tables 3-1, 3-2, and 3-3 that are not identified in Tables 4-1 through 4-6, 4-8 and 4-9. Explain how each of these aging mechanisms will be managed and revise MRP-227, if appropriate. If the aging management review (AMR) line items that were previously provided to update the Generic Aging Lessons Learned Report in conjunction with the staff's review of MRP-277 need to be revised, provide recommended changes to the AMR line items.

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 not to be either a Primary nor an Expansion component. As a result of the further evaluations, such components would not be identified in Tables 4-1 to 4-6 as either a Primary or an Expansion component, but could be identified in Tables 4-8 or 4-9 as a component covered by Existing programs. The following examples given in RAI 4-17 (in-core instrumentation thimble tubes in CE plants and control rod guide tube support pins in Westinghouse plants, etc.) are typical of this evaluation process.

In-Core Instrumentation Thimble Tubes in CE Plants. The Zircaloy-4 in-core instrumentation thimble tubes in CE plants were originally screened in for wear in MRP-191, primarily as the result of operating experience at San Onofre Units 2 and 3 and Waterford Unit 3. However, the further evaluation documented in Section 4.1.7 of MRP-

232 pointed out that modifications to the fuel alignment plate to alter the flow conditions in the vicinity of the entry point of the thimble tubes into the plate resolved the issue. Therefore, wear caused by flow-induced vibration is not expected to challenge the integrity of these components in the future. However, although irradiation-induced growth of zirconium alloys was not explicitly identified in MRP-175 as an age-related degradation mechanism to be evaluated as part of the screening process, it has been observed that irradiation-induced growth in the axial direction of the thimble tubes has reduced the clearance between the thimble nose and the bottom of the fuel assembly. As a result, some plants have observed that the thimbles were being compressed when the upper internals structure was replaced after fuel reload. Ten plants have taken some action to address the issue, and six of these plants have replaced the thimble tube assemblies with modified designs that are shorter in length to accommodate expected irradiation-induced growth in the future. Two additional plants have replacement designs in fabrication and have made preparations to install the replacement thimbles in an upcoming outage. The remaining two plants have not yet begun preparations for a full replacement of the thimble tubes, but have instead taken the intermediate step of raising the thimble support plate to accommodate additional axial growth. All of the affected plants will likely have replaced their thimble tubes prior to license extension. Westinghouse has provided plant-specific evaluations of the projected growth of the original thimble tubes, as well as recommendations for timing the replacement of these thimble tubes based on calculated clearances at the bottom of the fuel assembly to the affected CE-designed plants.

Because of the actions already carried out and the anticipated actions in the future, the management of this issue has been placed in the Existing Programs category.

Control Rod Guide Tube Support Pins in Westinghouse Plants. The control rod guide tube support pins in Westinghouse plants that were originally manufactured from Alloy X-750 material were screened in for primary water stress corrosion cracking (PWSCC), as well as wear and fatigue. The further evaluation of these components is documented in Section 4.2.5.2 of MRP-232. That further evaluation pointed out that failure of the guide tube support pins does not challenge safe plant operation, nor does such failure compromise control rod functionality. However, failure of guide tube support pins can result in a significant loose parts issue for the plant. In order to address this issue, some utilities have opted to replace guide tube support pins with a Type 316 cold-worked stainless steel support pin, others have opted to perform ultrasonic inspections to determine degradation, while some have preferred to take no action at this time. All domestic Westinghouse plants should or will have existing programs to replace or monitor guide tube support pins. As a result of past, current, and anticipated future actions, the management of this issue has also been placed in the Existing Programs category.