July 11, 2011

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

SUBJECT: REVISION 1 TO REQUEST FOR ADDITIONAL INFORMATION ON WCAP-17096-NP, REVISION 2, “REACTOR INTERNALS ACCEPTANCE CRITERIA METHODOLOGY AND DATA REQUIREMENTS” (TAC NO. ME4200)

Dear Mr. Wilmshurst:

By letter dated May 19, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession Number ML111160066), I requested that the Electric Power Research Institute (EPRI) respond to a request for additional information (RAI) concerning Topical Report (TR) WCAP-17096-NP, Revision 2, “Reactor Internals Acceptance Criteria Methodology and Data Requirements” (ADAMS Accession Number ML101460157). Since that request was made, the NRC staff has completed its review and safety evaluation for EPRI’s TR “Materials Reliability Program: Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines (MRP-227-Rev. 0)” (ADAMS Accession Number ML090160206), which is referenced in the WCAP. The changes made to the final version of the SE for TR MRP-227 requires the NRC to update the RAIs for TR WCAP-17096-NP, Revision 2.

The enclosure contains the updated RAIs for TR WCAP-17096-NP, Revision 2. The NRC requests your response to the enclosed Request for Additional Information (RAI) be submitted to the NRC Document Processing Center by Monday, August 15, 2011.

If you have any questions regarding this matter, 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 No. 669

Enclosure: RAI questions

July 11, 2011

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REVISION 1 to REQUEST FOR ADDITIONAL INFORMATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

WCAP-17096-NP, REVISION 2, "REACTOR INTERNALS ACCEPTANCE CRITERIA METHODOLOGY AND DATA REQUIREMENTS"

ELECTRIC POWER RESEARCH INSTITUTE

PROJECT NO. 669

The Electric Power Research Institute (EPRI) should revise the WCAP-17096-NP, Revision 2 report (referred to later as the WCAP report) in accordance with the safety evaluation (SE) on MRP-227-Rev. 0, "Materials Reliability Program: Pressure Water Reactor (PWR) Internals Inspection and Evaluation Guidelines (MRP-227-Rev. 0)."

SECTION 1 TO SECTION 5

RAI 1
It is stated in Section 2.0 of the WCAP report under "Evaluation Methodology" that the evaluation methodology includes "[a]n engineering basis for repair/replacement/mitigation options." Mitigation and repair are also included on the flow charts in Appendixes D and F. However, it is not discussed in the evaluation sections for the individual components in Appendixes C and E other than to indicate (for certain components) that if the acceptance criteria are not met the components require mitigation. Clarify that detailed recommendations on mitigation and repair options are outside the scope of this report. Also, confirm whether this information will be addressed in a future WCAP report.

RAI 2
It is stated in the first paragraph of Section 3.7 of the WCAP report, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," that, "[t]hree Expansion component items in the B&W [Babcock & Wilcox] designs have been designated for resolution by analysis." It is further stated in the second paragraph of Section 3.7 that, "[a]cceptance criteria for these three "Expansion" inspection category components are not included in this task." Assess the impact of leaving these three Expansion components outside the WCAP's inspection and evaluation (I&E) plan. Further, the staff found a discrepancy between the number of the "Expansion" inspection category components described in Section 3.7 (i.e., 3) and the number of "Expansion" inspection category components suggested on pages A-45, A-47, A-51, and A-53 of Appendix A to the WCAP report (i.e., 4). Clarification required.

RAI 3
Section 4.0 of the WCAP report states, in part, that although there are superficial differences in the format in which the results are presented by each vendor, the underlying structure of the information is the same. However, the staff notes that for the B&W reactor vessel internals (RVI) components, the information contains a heading, "What observations trigger examination into the Expansion category?" In some cases, the information under this heading provides additional definition beyond what was provided by the MRP-227-Rev. 0 report with respect to when inspection of an "Expansion" inspection category component is triggered. This

information is not included for the Combustion Engineering (CE) and Westinghouse RVI components. Revise Appendices C and E to include this information regarding what observations would trigger examination into the "Expansion" inspection category for CE and Westinghouse RVI components.

APPENDIX A AND APPENDIX B (B&W PLANTS)

RAI 4
For component items throughout Appendix A, it is mentioned frequently under "Methodology and Data Requirements," "perform analysis to…," or, "a … analysis is to be performed." Each of these analyses will provide a basis for the acceptance criterion for the examination results and will justify the adequacy of the proposed examination frequency for an RVI component. Provide a schedule for completion of all the analyses which are stated in the WCAP report as "to be performed" for the various RVI components and discuss actions to be taken if results from one or more analyses do not support the proposed I&E guidelines of a specific RVI component.

RAI 5
Pages A-1 to A-4 address core clamping items. The staff requests the following:

1. Page A-1 suggests that interference measurement to the nearest 0.001 inch can be achieved by way of "[d]etermination of differential height of top of plenum rib pads to reactor seating surface, with plenum in reactor vessel." Provide a schematic, demonstrating that this kind of accuracy is achievable through the physical measurements described above.
2. It is stated under "Methodology and Data Requirements" that "[t]he acceptance criterion is based on engineering judgment, and is defined as a reduction of no greater than 0.004 inch compared to the original as-built data." Discuss your engineering judgment. Demonstrate that after reduction of interference fit by 0.004 inch, there is still enough clamping load at operational conditions to prevent core movement due to the flow uplift loading and horizontal forces developed by various pump combinations.
3. It was stated under "Methodology and Data Requirement" that "[t]he general methodology to be used for VT-3 [general condition monitoring visual examination] acceptance criteria for these component items will be development of an NDE [nondestructive examination] inspection standard that contains examples of acceptable and unacceptable visual indications and mockups for the VT-3 inspection of wear." Provide the schedule for the VT-3 acceptance criteria development.

RAI 6
Page A-8 for core support shield (CSS) vent valve discs states that, "[c]onfirmed evidence of relevant conditions (damage, grossly cracked, or fractured material) in two or more vent valve disks shall require expansion to the control rod guide tube (CRGT) spacer castings by the completion of the next refueling outage." Explain why a plant could wait for two or more vent valve disks to be damaged before inspecting the CRGT spacer castings.

RAI 7
Pages A-9 to A-11 address CSS vent valve retaining rings and disc shafts. The staff requests the following:

1. Page A-9 states that, "[a]lthough there have been instances of failures of precipitation-hardenable materials in other applications, there is no known history of OE [operating

experience] identifying cracking in PWR reactor vessel internals applications." Clarify whether this history of no cracking is due to a lack of inspections performed on RVI components of precipitation-hardenable materials and discuss the relevance of these "other applications" to the current situation in terms of their differences or similarities in the component stress and environment.

2. Page A-10 states that, "[m]anufacturing and material data need to be identified to determine chemical composition and an assessment of the actual susceptibility to thermal aging embrittlement," for CSS vent valve top retaining ring, bottom retaining ring, and disk shaft or hinge pin. Confirm that the same action will be taken for the CSS cast outlet nozzles (Pages A-5 and A-6) and the CSS vent valve disks (Pages A-7 and A-8), consistent with the plant-specific action items in all license renewal safety evaluation reports.

<u>RAI 8</u>

Pages A-12 to A-16 address upper core barrel (UCB) bolts and locking devices. The staff requests the following:

1. Page A-14 states that, "[a] change of no more than 20 % in stiffness when subjected to LOCA [loss-of-coolant accident] loads is acceptable." Provide information regarding the establishment of the acceptance criterion of less than 20% change in overall core barrel stiffness. Since the change of the fundamental frequency of the structure is mentioned on Page A-14, confirm whether UCB dynamics was considered in determining this criterion? If yes, in what way?
2. Page A-14 states, "[a]n evaluation of joint stability (or, openness) is also to be performed." Define joint stability and openness in terms of parameters that can be calculated or measured. Confirm that this definition remains true throughout this report.
3. Page A-14 also states the need to, "[i]ncorporate the effect of future UCB bolt failure into the operability evaluation and re-inspection requirement." Elaborate on the operability evaluation and re-inspection requirement considering future UCB bolt failure. How is operability related to the overall core barrel stiffness, joint stability, and openness mentioned in the above two bullets?
4. Page A-15 states that, "[i]f inspection results indicate no relevant indications of failure and calculated peak stresses are below the bolt material yield strength, SCC is not expected to initiate and an inspection during the next ASME [American Society of Mechanical Engineers] Code B-N-3 10-year ISI [inservice inspection] interval is judged adequate." Provide industry operating experience showing that (1) all failed UCB bolts have calculated peak stresses above the bolt material yield strength, and (2) once a crack was initiated, it did not grow to a critical size in 10 years.
5. Page A-16 states that cracking observed in 10% (i.e., 12) of the UCB bolts would trigger examination into the "Expansion" inspection category. Demonstrate that the "Expansion" inspection category components are under a much less aggressive degradation conditions such that you can wait for 12 UCB bolts to fail to take action on "Expansion" inspection category components.

<u>RAI 9</u>

Pages A-17 to A-20 address lower core barrel (LCB) bolts and locking devices. Discuss the applicability of the response to RAI 8 (except for the first item) for the UCB bolts and locking devices to the LCB bolts and locking devices.

RAI 10
Pages A-21 to A-24 address core barrel assembly baffle-to-former (BF) bolts. It is stated under "Existing Documentation" that "[p]ast B&WOG work has determined the minimum number of bolts for safe shutdown, but not for operation." Explain how this analysis was used to establish the criterion for conducting examination into the "Expansion" inspection category, i.e., 5% (or 40 BF bolts) with observed cracking or greater than 25% of the bolts on a single former plate.

RAI 11
Pages A-25 to A-27 address core barrel assembly baffle plates. The staff requests the following:

1. Page A-26 states under "Methodology and Data Requirements" the need to "[d]etermine the expected crack opening displacement (COD) for development of the inspection standard." Discuss how the COD is going to be used in the inspection and in a flaw evaluation handbook with the calculated critical crack size.
2. Page A-27 states under "What Observations Trigger Examination into Expansion Category," "[i]f a VT-3 examination is possible, it is required by completion of the next refueling outage." Discuss conditions under which a VT-3 examination would not be possible for the core barrel and former plates. Propose a remedy if the coverage is less than the minimum examination coverage requirement of 75% specified as Condition 4 in the SE for the MRP-227-Rev. 0 report.

RAI 12
Pages A-35 to A-36 address dowel-to-upper grid fuel assembly support pad welds or dowel-to-lower grid fuel assembly support pad welds. Possible examination outcomes include one or several support pads are misaligned or missing. If this happened, clarify whether loading of the fuel into the core will become a problem. How do you define the severity of the misalignment such that and beyond which loading of the fuel into the right position becomes a problem? Fully explain.

RAI 13
Pages A-39 to A-41 address upper thermal shield (UTS) bolts and their locking devices. Provide a justification for not addressing the effect of changing the fundamental frequency of the UTS due to failed UTS bolts or their locking devices, causing the UTS to be not firmly attached to the core barrel at certain places. Further, page A-41 of the WCAP report requires incorporation of the effect of future UTS bolt failure into the operability and re-inspection requirement. Define the number or percentage of failed UTS bolts that would trigger an operability evaluation and elaborate on the contents of the re-inspection requirement (e.g., inspection method, scope, frequency, etc.).

RAI 14
Pages A-42 to A-44 address surveillance specimen holder tube (SSHT) studs/nuts or bolts and their locking devices. Define the degree of the SSHT stud/nut or bolt failure that would trigger an operability evaluation.

RAI 15
Pages A-45 to A-46 address core barrel cylinders. The staff requests the following:

1. Since the "Examination Coverage" is designated as "Inaccessible" and no inspection is proposed, specify the condition that would trigger a need for a justification by evaluation or by replacement and discuss, consistent with operating experience, the situations

which would require disassembly of RVI components to make inspection of the core barrel cylinder possible.

2. The WCAP report states under "Observable Effects" that "[t]he core barrel upper flange-to-core barrel weld and upper HAZ [heat-affected zone] area is partially accessible and could potentially be VT-3 examined." Provide guidelines for inspecting this partially accessible area and specify the level of degradation that can be determined from this partially accessible area such that a meaningful operability evaluation can be made for the core barrel cylinder to operate for at least one cycle.

RAI 16
Except for minor differences, RAI 15 also applies to the former plates. Therefore, using your response to RAI 15 for core barrel cylinders as an example, provide similar responses for the former plates.

RAI 17
Pages A-49 to A-50 address core baffle-to-baffle (BB) bolts. It is stated under the table column "Examination Coverage" that "Acceptable examination technique not currently available" for the internal BB bolts and "Inaccessible" for the external BB bolts. Explain how the licensee can evaluate the degradation of the BB bolts using the finite element method (FEM) analysis results (i.e., an acceptable failure pattern or the number of failure bolts allowed) developed by the MRP if the WCAP report proposed no inspection for these BB bolts to verify the plant's current failure pattern or the number of failure bolts. Provide the basis for the BB bolt failure locations assumed in the FEM analysis and explain why the FEM results are bounding.

RAI 18
Except for minor differences, RAI 17 also applies to the core barrel-to-former (CF) bolts discussed in pages A-51 to A-52. Therefore, using your response to RAI 17 for core BB bolts as an example, provide similar responses for the CF bolts.

RAI 19
Pages A-56 to A-58 address lower grid shock pad bolts and locking devices. The staff requests the following:

1. It is stated under "Component Item Function" that "[t]he function of the lower grid shock pad bolts is to fasten the shock pads to the lower grid assembly. Shock pads must be in place to carry accidental core drop loads. The bolts do not function to carry the core drop load, but serve to hold the pad in place." Substantiate the above statement by providing a sketch of the pad and the bolt, which shows the direction of the core drop load and the location of the impact.
2. It is stated under "Methodology and Data Requirements" that "[i]f two bolts on any individual shock pad are identified with relevant indications, [a] structural evaluation is to be performed to determine if remaining pads can carry the core drop load or if the load can be carried without the shock pad in place." Provide detailed load distribution for the case when the load can be carried without the shock pads in place.

RAI 20
Pages A-62 to A-64 address the flow distributor bolts and their locking devices. It is stated under "Existing Documentation" that "[a] generic flow distributor bolt stress analysis (for all units except TMI-1 [Three Mile Island, Unit 1]) was developed for the MRP reactor internals project in 2007." Explain how the results and conclusions from this existing generic flow distributor bolt

stress analysis will be utilized in addressing the acceptance criteria and methodology and data requirements for this RVI component.

RAI 21
The flowchart on page B-5 for core support shield vent valve retaining rings and disc shaft indicates that, if no NDE condition is confirmed, "[n]o subsequent inspection is required unless triggered by new inspection results [upper right box of the flowchart]." This is inconsistent with the inspection information on Page A-9 for these components, which places no conditions on the first and subsequent visual examinations. Clarification required. Further, the no subsequent inspection option is unique among all "Primary" inspection category component items. Needs justification.

RAI 22
The flowchart on page B-12 for incore monitoring instrumentation guide tube spiders and welds indicates that, if relevant conditions exist, the visual examination for discontinuities and imperfections (VT-1) may be used for better characterization. This is inconsistent with the inspection information on Page A-33 for the same components, which also includes eddy current testing (ET) and UT as possible inspection methods. Clarification required.

RAI 23
The flowchart on page B-19 for BB bolts (A-49) and CF bolts (A-51) mentioned a new inspection technique. Provide details on this new inspection technique. Has the new inspection technique been given serious thought or is it under development? How much coverage can this new inspection technique achieve? Confirm that the new inspection technique applies to the locking devices for external BB and CF bolts (the flowchart on Page B-20).

RAI 24
Consistent with the conditions specified in the SE for the MRP-227-Rev. 0 report, revise the inspection category for the following B&W RVI components. Accordingly, for those components not previously in the "Primary" or "Expansion" inspection category, revise Appendix A of the WCAP report to include the acceptance criteria methodology and data requirements and Appendix B of the WCAP report to include the flowchart for these components. For those components being changed from the "Expansion" inspection category to the "Primary" inspection category, modify Appendix A and B as necessary to reflect this change.

1. Flow Distributor-to-Shell Forging Bolts: Condition 3 of the SE for the MRP-227-Rev. 0 report requires the report to include this component in B&W designed reactors in the "Primary" inspection category.
2. Core Support Structure Upper Flange: Action Item 4 of the SE for the MRP-227-Rev. 0 report requires the report to include this component in B&W designed reactors in the "Primary" inspection category if this component has not been stress relieved.

APPENDIX C AND APPENDIX D (COMBUSTION ENGINEERING PLANTS), APPENDIX E AND APPENDIX F (WESTINGHOUSE PLANTS)

RAI 25
The staff requests the following information related to Appendix E, Item W-ID: 1 Control Rod Guide Tube Assembly – Guide Plates (Cards) (pp. E-2 to E-3):

1. The calculation of wear life determines the wear fraction at Stage A wear. The different stages are defined as follows: "During Stage A, one should be able to observe a flat, unworn

surface on slots in inner hole. Upon transition from Stage A to Stage B, there is no observable slot. Sharp point observed where wear area intersects inner surface of guide card." V_a is the wear volume at the transition from Stage A to Stage B. V_B is the wear volume at point where width of wear area at guide tube inner surface is equivalent to control rod diameter. The wear fraction is $f = V_a/V_b$. The remaining life is $T = (1/f - 1)T_{cur}$, where T_{cur} is the current operating time.

   a. What would the typical wear fraction be at the transition from Stage A to Stage B?
   b. How is Stage A defined quantitatively, i.e., a certain "f" or a certain size ligament before the slot starts to widen?

2. The methodology for determining remaining wear life does not provide any margin. Wear could have accelerated due to a change in operation, such as a higher flow rate due to a power uprate. Justify not requiring some margin in terms of the inspection interval, such as requiring re-inspection at some fraction of the calculated "T".

<u>RAI 26</u>

The staff requests the following information related to Appendix E Item W-ID: 2 Control Rod Guide Tube (CRGT) Lower Flange Welds (pp. E-4 to E-5):

1. Under "Methodology" is stated that allowable load on CRGT assembly is determined by empirical testing. Has this empirical testing been done or will the testing be done for each plant prior to performing this evaluation?
2. Under "Data Requirements" is stated, "[f]inite element model of lower CRGT assembly to evaluate weld failures calibrated to benchmark data." Is the "benchmark data" the data from the empirical test?
3. Item #3 of the analysis procedure included in the "Methodology" section for the CRGT Lower Flange Welds states "Calibrate finite element analysis (FEA) model and boundary constraints against design basis assumptions." Clarify whether the "design basis assumptions" include LOCA and safe shutdown earthquake (SSE) loads.
4. The inspection coverage of the CRGT lower flange welds is 100% of outer (accessible) CRGT lower flange weld surfaces. If cracked welds are observed, how is the structural integrity of the CRGT lower flange welds that are inaccessible for inspection evaluated?
5. How is the re-inspection interval determined if some welds are found to be cracked?

<u>RAI 27</u>

The staff requests the following information related to the items listed below:

- Appendix C Item CE-ID: 1.2 Core Shroud Assembly (Bolted) Core Support Column Bolts (pp. C-6 to C-7)
- Appendix C Item CE-ID: 6.3 Lower Support Structure – Lower Support Column Welds (pp. C-26 to C-27)
- Appendix E Item W-ID: 2.1 Lower Support Assembly – Lower Support Column Bodies (Cast) (pp. E-6 to E-7)
- Appendix E Item W-ID: 3.2 Lower Support Assembly – Lower Support Column Bodies (Non-Cast) (pp. E-14 to E-15)
- Appendix E Item W-ID: 5.2 Lower Support Assembly – Lower Support Column Bolts (pp. E-22 to E-23)

All the components listed above use an identical acceptance criterion of $N_f < (N-N_{req})/2$ where:
N = # of components,
N_f = # of observed failed components
N_{req} = # of columns in relevant minimum pattern

1. The bases for the assumption in the analysis procedures for the items listed above that the number of failures in the next 10-year interval will be equal to the number observed to date and the acceptance criterion are not clear.
   a. Explain how the acceptance criterion for failed components was determined.
   b. Discuss whether a certain distribution of failures versus effective full power years (EFPY) of operation was assumed (for example, linear, normal, Weibull, etc.)
   c. Justify the distribution chosen using operating experience or other considerations.
2. Since, in the minimum acceptable pattern of core support columns and bolts, the location as well as the number of uncracked columns and bolts may be important, a simple numerical margin may not be appropriate. Discuss whether the acceptance criterion needs to consider the location of the additional uncracked columns and bolts that constitute the margin.

RAI 28

The staff requests the following information related to Appendix E Item W-ID: 2.2 Bottom Mounted Instrumentation System – Bottom Mounted Instrumentation (BMI) Column Bodies (pp. E-8 to E-9):

1. The item is categorized as an “Expansion” inspection category component in the MRP-227-Rev. 0 report. However, under inspection, the required coverage is specified as 100% of BMI column bodies for which difficulty is detected during flux thimble insertion/withdrawal. Clarify if BMI column bodies would be inspected when cracking is found in CRGT flange welds, or only in the case that difficulty in inserting a flux thimble occurs.
2. If BMI column bodies are required to be inspected as an “Expansion” inspection category component, as opposed to inspection due to difficulty in inserting/removing the flux thimble, confirm that the acceptable minimum population of BMI column bodies to be inspected is at least 75% of the total number of BMI column bodies consistent with Condition 3 of the SE for the MRP-227-Rev. 0 report.
3. If difficulty is encountered inserting flux thimbles at the beginning or end of the refueling outage (RFO), is inspection of all BMI bodies required? If the difficulty in inserting the flux thimble occurs at the end of the RFO, can the inspection be deferred until the subsequent RFO?

RAI 29

The staff requests the following information related to the items listed below:

- Appendix C Item CE-ID: 6 Core Support Barrel Assembly Upper (Core Support Barrel) Flange Weld (pp. C-20 to C-22)
- Appendix E Item W-ID: 3 Core Barrel Assembly – Upper Core Barrel Flange Weld (pp. E-10 to E-11)

1. Under the “Analysis” procedure for the above items, Option 1, Step 1 states “Determine stress distribution through core support barrel thickness for normal operating conditions (expect peak stress at vessel outer diameter (OD)).” Should this stress analysis to determine the highest stress surface be completed before the surface to be visually inspected (OD or inner diameter (ID)) is selected?
2. Provide details on how the maximum constrained crack length, L_c, is determined. Is the stress distribution in the weld determined by an FEM analysis? Does this pertain to axial cracks, circumferential cracks, or both?
3. For these welds, MRP-227-Rev. 0 specifies an EVT-1 examination of 100% of the accessible surfaces. MRP-227-Rev. 0 does not specify that these examinations may be conducted from only one side of the weld (i.e., OD or ID). However, the analysis procedures of WCAP-17096-NP appear to offer the option of performing an examination from either the

OD or the ID. If the examination is performed from the ID, and flaws are observed less than the critical flaw length, L_c, the analysis procedure requires a supplementary visual examination of the OD or a UT examination to determine if the crack is OD initiated. However, if no flaws are observed in the ID examination, no supplementary examination is required. How can it be assured that no structurally significant OD-initiated non-throughwall cracks are present if no cracks or flaws are observed on the ID surface and no supplementary examination is performed?

4. The flowcharts on page D-10 and F-6 show that either UT examination or OD visual inspection may be used when a crack/flaw is observed during an EVT-1 examination conducted from the weld ID. Therefore Option 2 and 3 described in the analysis procedures for CE-ID:6 and W-ID:3 are actually subsets of the same option. The staff recommends that EPRI revise the analysis procedure for CE-ID:6 and W-ID:3 to be consistent with the flowcharts.
5. Step 6 of the methodology refers to an applied stress intensity factor (K) limit of less than 20 ksi√in below which the crack is acceptable. What is the basis for accepting a flaw with an applied K less than 20 ksi√in? What is the fracture mechanics methodology, i.e., linear elastic fracture mechanics (LEFM), elastic plastic fracture mechanics (EPFM), or limit load, used in this evaluation? What is the applicable neutron fluence range? Is it consistent with the MRP-227-Rev. 0 report, Section 6 recommendations that for neutron exposures of 5 - 15 displacements per atom (dpa), LEFM analyses should be used with a K_{Ic} of 55 MPa√m (50 ksi√in); and for components receiving 15 dpa or greater, a K_{Ic} of 38 MPa√m (34.6 ksi√in).
6. Are there neutron fluence limitations on the use of the recommended evaluation procedure?
7. Discuss how multiple cracks in close proximity would be handled.
8. If it cannot be demonstrated that the observed flaws/cracks are not growing, what corrective actions must be taken? If re-inspection on an interval shorter than 10 years is proposed as a corrective action, provide also how the shorter re-inspection interval is determined.

<u>RAI 30</u>

The staff requests the following information related to the items listed below:

- Appendix C, Item CE-ID: 6.2 Core Support Barrel Assembly – Remaining Core Barrel Assembly Welds (pp. C-24 to C-25)
- Appendix E, Item W-ID: 3.1 Core Barrel Assembly – Core Barrel Flange, Core Barrel Outlet Nozzles, and Lower Core Barrel Flange Weld (pp. E-12 to E-13)

1. For these items, under “Methodology # 6,” it states, “[u]se appropriate crack growth rate models (stress corrosion cracking (SCC) or irradiation assisted stress corrosion cracking (IASCC) and fatigue) to estimate crack growth rate.” The MRP-227-Rev. 0 report, Section 6 recommends the use of the boiling water reactor (BWR) hydrogen water chemistry (HWC) crack growth rate model until generic curves are established for IASCC and SCC crack growth in the PWR environment, and refers to the MRP-211 report, “Materials Reliability Program: PWR Internals Age-Related Material Properties, Degradation Mechanisms, Models, and Basis Data - State of Knowledge,” as the source of this model.
    a. Identify the crack growth rate model(s) to be used for IASCC and SCC.
    b. Confirm that the BWR HWC crack growth rate model is being used as recommended by the MRP-227-Rev. 0 report. If a different model will be used, provide the model.
2. Provide the crack growth rate model to be used for fatigue crack growth.

RAI 31

The staff requests the following information related to the items listed below:

- Appendix C, Item CE-ID: 6.2 Core Support Barrel Assembly – Remaining Core Barrel Assembly Welds (pp. C-24 to C-25)
- Appendix E, Item W-ID: 3.1 Core Barrel Flange, Core Barrel Outlet Nozzles, and Lower Core Barrel Flange Weld (pp. E-12 to E-13)

1. For the subject components, the procedure given in the methodology section recommends the use of a fracture toughness K_{IC} = 150 ksi√in for low neutron fluence regions. Use of a K_{IC} fracture toughness implies the use of LEFM. However, for lower neutron fluence regions, defined as receiving a neutron exposure less than 5 dpa, the MRP-227-Rev. 0 report, Section 6 recommends the use of EPFM. Provide the basis for using a toughness of K_{IC} = 150 ksi√in, and confirm whether LEFM or EPFM will be used for evaluating flaws in the subject components.
2. The acceptance criteria for the above items are based on the crack sizes being "explainable by known crack growth rate laws." Elaborate on "crack sizes being explainable by known crack growth rate laws," and explain how initiation time or incubation period is accounted for in determining that a crack size is explainable.
3. Discuss how multiple cracks in close proximity would be handled.

RAI 32

The staff requests the following information related to the items listed below:

- Appendix C, Item CE-ID:1 Core Shroud Assembly (Bolted) – Core Shroud Bolts (pp. C-2 to C-3),
- Appendix C, Item CE-ID: 1.1 Core Shroud Assembly (Bolted) – Barrel-Shroud Bolts (pp. C-4 to C-5)
- Appendix E, Item W-ID: 5 Baffle-Former Assembly – Baffle-Former Bolts (pp. E-18 to E-19)
- Appendix E, Item W-ID: 5.1 Core Barrel Assembly – Barrel-Former Bolts (pp. E-20 to E-21)

1. The "Analysis" section of the evaluations for the above items defines a minimum required number of bolts N_{req}
    a. Describe how the required number of bolts N_{req} is determined.
    b. If the determination of N_{req} is based on a generic methodology previously submitted to the NRC, provide the reference for the methodology.
    c. If not, describe the general methodology to be used to determine the required number and distribution of bolts, providing an equivalent level of detail to Appendix A, pages A-21 through A-24.
2. The acceptance criterion at the first inspection is no more than 50% of the initial margin consumed. This is supposed to ensure that the margin will not be exceeded during the 10-year re-inspection interval required by Condition 5 of the SE for the MRP-227-Rev. 0 report. The MRP-227-Rev. 0 report recommended the initial inspection at 25 - 35 EFPY. This acceptance criterion seems to be based on a linear failure distribution for the bolts. How was the acceptable margin for failed bolts determined? Discuss whether a certain distribution of failures versus EFPY of operation was assumed (for example, linear, normal, Weibull, etc.). Justify the distribution chosen using operating experience or other considerations.
3. Since in the minimum bolting pattern the location as well as the number of remaining bolts is important, a simple numerical margin may not be appropriate. If the minimum number of bolts also requires a certain acceptable pattern or distribution, discuss whether this needs to be considered when determining if the remaining margin is adequate.

4. The MRP-227-Rev. 0 report specifies for examination coverage of baffle-former bolts, 100% of the accessible bolts. In its October 29, 2010, response to NRC staff RAI 4-8 related to the MRP-227-Rev. 0 report, the MRP also proposed to revise the MRP-227-Rev. 0 report to require that, when addressing a set of like components (e.g., bolting), the inspection examines a minimum sample size of 75% of the total population of like components.
   a. If some bolts are not inspected due to inaccessibility, how is the number of failed bolts N_f in the entire bolt population determined?
   b. What assumptions are made regarding the inaccessible bolts?
5. There is certain minimum flaw size (30% per the MRP-227-Rev. 0 report), as a percentage of the bolt cross section, that can be detected by the UT technique used to examine baffle-former and barrel-former bolts. For high neutron fluence bolts, the allowable flaw size could be below the detectability limit of the UT examination. How are the limitations of the UT examination technique, in terms of the minimum detectable flaw size, accounted for in determining the number of bolts required, and/or the required margin?
6. Are bolts determined to be failed via the UT examination required to be replaced?
7. Has consideration been given to requiring removal of some failed bolts for failure analysis to confirm the failure mechanism?
8. Provide a summary of the operating experience related to inspection of Westinghouse baffle-former and barrel-former bolts, and CE core shroud bolts and barrel shroud bolts. The summary should focus on augmented UT examinations. Discuss how the operating experience is factored into the evaluation and acceptance criteria for the bolts listed above.
9. Condition 5 of the staff's draft SE of the MRP-227-Rev. 0 report required a 10 year inspection frequency for baffle-former bolts of Westinghouse-designed reactors and core shroud bolts (versus 10-15 years specified in the MPR-227 report) in CE-designed reactors unless an applicant/licensee provides an evaluation for NRC staff approval which justifies a longer interval between inspections. The staff requests that EPRI modify the WCAP to reflect this change.

RAI 33

This is a continuation of RAI 32 reflecting the staff's concern on loss of fracture toughness due to irradiation embrittlement. Due to the detectability limitations on the UT examination technique for the RVI bolting, it is possible that bolts not identified to be cracked could have cracklike flaws of less than 30% of the cross sectional area. For this reason, the staff's SE for the WCAP-15029-P-A report, "Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions," recommended the following:

- In using irradiated tensile strength for calculating safety margins, the bolt load-bearing cross-section area should be reduced based on the sensitivity of the ISI methods and the potential for degraded bolts to escape detection by ISI.
- Since the available data for irradiated stainless steel indicates that the ductility (and presumably the fracture toughness) of irradiated bolts is severely degraded, a fracture mechanics approach would be necessary to demonstrate that the degraded bolts exhibit adequate toughness with a postulated flaw size undetected by ISI.

The following limitations on the use of the WCAP-15029 methodology were imposed by the staff:

- The methodology should not be used to assess existing bolting without demonstration of adequate conservatism in projected bolting material properties (i.e., yield and ultimate strength) to ensure that sufficient ductility is present in existing irradiated stainless steel bolting materials.

- The use of the methodology for existing irradiated stainless steel bolting should account for limitations in available ISI methods with regard to the probability of detection characteristics.

The staff therefore requests the following information for the bolts listed in RAI 32:

1. Discuss how irradiation embrittlement is accounted for in the determination of the minimum required bolting pattern.
2. Discuss how potential undetected flaws in bolts are accounted for in the determination of the minimum required bolting pattern.
3. How are the lessons learned from previous baffle-former bolt inspections incorporated into the acceptance criteria and evaluation methodology for the baffle-former bolts and other RVI bolts subject to UT examination?

RAI 34

The staff requests the following information related to the items listed below:

- Appendix C, Item CE-ID: 2 Core Shroud Assembly (Welded) (pp. C-8 to C-9)
- Appendix C Item CE-ID: 2.1 Core Shroud Assembly (Welded) – Remaining Axial Welds (pp. C-10 to C-11)
- Appendix C, Item CE-ID: 3 Core Shroud Assembly (Welded) Shroud Plates (pp. C-12 to C-13)
- Appendix C, Item CE-ID: 3.1 Core Shroud Assembly (Welded) – Remaining Axial Welds, Ribs, and Rings (pp. C-14 to C-15)

1. The analysis procedures direct the evaluator to construct models for fatigue and IASCC crack growth rates as a function of K. No specific crack growth rate model is identified. The MRP-227-Rev. 0 report, Section 6 recommends the use of the BWR HWC crack growth rate model until generic curves are established for IASCC and SCC crack growth in the PWR environment, and refers to MRP-211 as the source of this model.
    a. Identify the specific crack growth rates to be used. Specify the applicable neutron fluence range for these crack growth rates.
    b. If using different crack growth rates than recommended by the MRP-227-Rev. 0 report, provide the reference and justification for the alternate crack growth rates.
2. The analysis procedures recommend using limiting fracture toughness, K_{Ic}, for highly irradiated material, for the center of core shroud location. However, the specific value of K_{IC} for highly irradiated material is not identified. The MRP-227-Rev. 0 report Section 6 recommends for neutron exposure greater than 5 dpa: LEFM analyses should be used with a limiting fracture toughness K_{Ic} = 55 MPa$\sqrt{m}$ (50 ksi$\sqrt{in}$) for exposure levels between 5 and 15 dpa, and with a limiting fracture toughness K_{Ic} = 38 MPa$\sqrt{m}$ (34.6 ksi$\sqrt{in}$) for exposure levels greater than 15 dpa.
    a. Identify the bounding K_{IC} for highly irradiated material (consistent with the MRP-227-Rev. 0 report, Section 6.0 or other).
    b. If a bounding K_{IC} different than that recommended by the MRP-227-Rev. 0 report is used, provide the reference and justify the use of the alternate K_{IC} values.
    c. Identify the neutron fluence thresholds for each bounding K_{Ic} value.
3. For neutron exposure less than 5 dpa, the MRP-227-Rev. 0 report recommends using EPFM rather the LEFM to evaluate flaws. Are there any portions of the core shroud assembly that could have neutron exposure below 5 dpa, for which use of EPFM techniques would be appropriate?
4. The analysis procedures for these items include the following: “Obtain stress intensity factor (K) solution corresponding to crack at comer with described loads.” This implies the crack is always a corner crack. Discuss the K solutions to be used if the crack is not a corner crack.

5. Discuss how multiple cracks in close proximity would be handled.

RAI 35
The staff requests the following information related to Appendix C, Item CE-ID:7 Core Support Barrel Assembly – Lower Flange Weld (pp. C-28 to C-29):

1. Under "Analysis" is recommended "Establish criteria for the highest probability surface. Demonstrate that a 1/4 thickness flaw of observed length will not grow through barrel wall in planned inspection interval." However, under "Analysis" #4, "Establish Criteria for Lower Probability Surface," a UT exam can be performed to demonstrate the flaw is limited to the initiating surface.
    a. What is the basis for assuming a 1/4 thickness flaw in the weld based on a visual examination only?
    b. If UT examination is feasible for this weld, if a flaw is visually detected on the highest probability surface, why not perform UT to determine the depth of the flaw observed on the highest probability surface, rather than assuming a 1/4t depth?
2. Under "Methodology and Data Requirements," stress corrosion cracking (SCC) crack growth rate curves are said to be needed as a backup. Under "Analysis" is stated that "Due to general concerns about SCC in structural welds, the same location has been listed as an expansion inspection that would be triggered by observation of cracking in the upper flange weld." However, no specific SCC crack growth rate models are mentioned.
    a. Identify the crack growth rate model(s) to be used for IASCC and SCC. Is the BWR HWC crack growth rate model being used as recommended by the MRP-227-Rev. 0 report?
    b. If a different model will be used, provide the model.
3. Discuss how multiple cracks in close proximity would be handled.

RAI 36
The following two primary components will only be inspected if the potential for fatigue cracking cannot be resolved by the time limited aging analysis (TLAA) process:

- Appendix C, Item CE-ID: 8 Lower Support Structure – Core Support Plate (pp. C-30 to C-31)
- Appendix C, Item CE-ID: 9 Upper Internals Assembly – Fuel Alignment Plate (pp. C-32)

However, the description of the analysis process is much more detailed for the core support plate. Discuss whether both components should use a similar analysis process.

RAI 37
The staff requests the following information related to CE-ID: 11 Lower Support Structure – Deep Beams (pp. C-35 to C-36):

1. Under "Analysis #1" is stated "[t]he grid structure of the lower core support in these plants precludes catastrophic failure initiated by a single crack." However, the methodology does not specifically address the case of multiple cracked beams. Demonstrate that the deep beam grid structure will maintain its structural integrity with a certain number of beams or welds completely failed.
2. Under "Analysis #2" is stated "[m]aximum acceptable crack size corresponds to projected remaining ligament = 0." A projected remaining ligament of zero would result in failure of the beam or weld. This provides no margin. Justify allowing a remaining ligament of zero for a beam or weld with respect to the structural integrity of the deep beam grid structure.

RAI 38

The components listed below are susceptible to IASCC:

- Appendix C Item CE-ID: 1.2 Core Shroud Assembly (Bolted) Core Support Column Bolts (pp. C-6 to C-7)
- Appendix E Item W-ID: 5.2 Lower Support Assembly – Lower Support Column Bolts (pp. E-22 to E-23)

However, the data requirements for evaluation of these bolts do not include the irradiated material properties. Other components in the lower internals that perform a core support function include the irradiated material properties among the data requirements, such as Appendix C Item CE-ID: 6.3 Lower Support Structure – Lower Support Column Welds, Appendix E Item W-ID: 2.1 Lower Support Assembly – Lower Support Column Bodies (Cast), and Appendix E Item W-ID: 3.2 Lower Support Assembly – Lower Support Column Bodies (Non-Cast).

Discuss whether the data requirements for these components should include the irradiated material properties.

RAI 39

The following item is subject to an EVT-1 examination:

Appendix C Item CE-ID: 6.1 Core Support Barrel Assembly – Lower Core Barrel Flange Weld (pp. C-22 to C-23)

An FEM analysis is proposed to determine the stress distribution across the weld and thus determines the most likely surface for crack initiation. Under “Analysis” #3 and #4, criteria are then proposed for the most and least likely surfaces. The criteria include a UT or visual examination to verify the flaw is limited to the initiating surface, and verifying the flaw growth will be self-limiting due to the stress distribution, or is not consistent with an actively growing crack.

1. The criteria of “Analysis” #3 and #4 do not appear to project the future crack growth, nor do they determine the acceptability of the crack at the end of the interval. Discuss how these objectives are accomplished.
2. Discuss why a different analysis approach is used for this item than for CE-ID: 6.2 Core Support Barrel Assembly – Remaining Core Barrel Assembly Welds.
3. Discuss how multiple cracks in close proximity would be handled.

RAI 40

The staff requests the following information related to Appendix C Item CE-ID: 5 Core Shroud Assembly (Welded) – Assembly (pp. C-18 to C-19).

1. Under “Failure Criteria,” one of the criteria is “Gap size implies peak shroud swelling > 5 % by volume.” However, according to the MRP-227-Rev. 0 report, 5% swelling is considered severe and can correlate with extremely low fracture toughness values. Should the failure criteria for gap size be correlated with a lower percentage of swelling in order to provide some margin before the onset of severe swelling?
2. Void swelling can only be detected or measured in the field by measuring the distortion of components or assemblies. Void swelling has been modeled for CE- designed RVI components and assemblies, such as the CE welded core shroud, as documented in the MRP-230 report, “Functionality Analysis for Westinghouse and Combustion Engineering Representative PWR Internals.” Will the measured swelling be compared to such modeling efforts to determine if the models accurately predict swelling in the RVI components? How will this be accomplished?
3. Discuss the relevance of previous failure modes and effects analyses (FMEA) and failure modes, effects and consequence analyses (FMECA) efforts, such as that documented in the

MRP-191 report, “Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Designs,” to the evaluation of Appendix C Items CE-ID: 4 Core Shroud Assembly (Bolted) – Assembly and CE-ID: 5 – Core Shroud Assembly (Welded) – Assembly.

RAI 41

The items listed below are susceptible to cracking due to fatigue and are only inspected if they cannot be qualified via a TLAA:

- Appendix C Item CE-ID: 7 Core Support Barrel Assembly – Lower Core Barrel Flange (pp. C-28 to C-29)
- Appendix C Item CE-ID: 8 Lower Support Structure – Core Support Plate (pp. C-30 to C-31)
- Appendix C Item CE-ID: 9 Upper Internals Assembly – Fuel Alignment Plate (p. C-32)
- Appendix C Item CE-ID: 11 Lower Support Structure – Deep Beams (p. C-35)

Under “Acceptance Criteria” the WCAP report states, “Acceptance criteria for TLAA related items are beyond the scope of the current project.” However, if the items are being inspected, it is because the fatigue life could not be qualified via a TLAA for the life of the plant. Therefore, it seems that the acceptance criteria for this evaluation should not be defined by the TLAA. The staff requests that EPRI discuss whether it is possible to define acceptance criteria based on ensuring structural integrity of the components until the next inspection.

RAI 42

A FMEA is proposed to determine the acceptance criteria for Appendix E Item W-ID: 4 Baffle-Former Assembly – Baffle-Edge Bolts (pp. E-16 to E-17). Per the MRP-227-Rev. 0 report, these bolts are a “Primary” inspection category component to be inspected via VT-3 examination for missing or broken locking devices, failed or missing bolts, or protruding bolts heads. The staff requests the following information:

1. Will be baffle-edge bolt FMEA be part of the overall baffle-former assembly FMEA performed for Appendix E Item W-ID: 6 Baffle Former Assembly – Assembly (pp. E-24 to E-25)?
2. Since failure of these bolts could lead to gaps between the baffle plates that could lead to baffle jetting, will gaps between plates observed as part of the baffle-former assembly visual VT-3 examination be considered in the FMEA?
3. Since UT examination is not being performed, bolts could be failed without any visible evidence if the locking device retains the bolt. How will this be accounted for in the FMEA?
4. Provide more detail on the process to be used for the FMEA(s) for Appendix E Item W-ID: 6 Baffle-Former Assembly – Assembly and Appendix E Item W-ID: 4 Baffle-Former Assembly – Baffle-Edge bolts.

RAI 43

- Appendix C Item CE-ID: 6.1 Core Support Barrel Assembly – Lower Core Barrel Flange (pp. C-22 to C-23)(MRP-227-Rev. 0 specifies 100% of accessible)
- Appendix C Item CE-ID: 6.2 Core Support Barrel Assembly – Remaining Core Barrel Assembly Welds (pp. C-24 to C-25)

For the components listed above, the examination method specified by MRP-227-Rev. 0 is an EVT-1 visual examination of all accessible welds and adjacent base metal. For CE-ID:6.2, the examination is one-sided. For CE-ID:6.1, a one-sided examination appears to be permissible since the flowchart on page D-11 specifies that observed flaws are assumed to be throughwall. These items are considered acceptable if no cracklike flaws are observed during the visual examination. The analysis procedure for CE-ID:6.1 includes an FEA to determine the stress distribution though the welds, but the analysis procedure for CE-ID:6.2 does not include this step. For these welds, discuss how it can be assured that no crack has initiated on the opposite (OD) surface if the examination of the ID surface reveals no flaws or cracks.

RAI 44

For Appendix E Item W-ID: 8 Thermal Shield Assembly – Thermal Shield Flexures (pp. E-28), if flawed flexures are observed, a structural analysis is proposed to demonstrate that the dynamic response of the thermal shield is unchanged when the flawed thermal shield flexure(s) are removed from the model.

1. What actions should be taken if the dynamic response of the thermal shield changes as a result of the failed flexures removal?
2. The acceptance criteria also state that any observation of a failed thermal shield flexure should lead to enhanced vigilance for fatigue and vibration monitoring systems. Elaborate on what this involves. How would the potential for fatigue be monitored? What would be recommended if there is no existing vibration monitoring system?

RAI 45

Condition 1 of the draft SE for the MRP-227-Rev. 0 report requires that several components classified as the "No Additional Measures" category be moved to the "Expansion" category due to the high consequence of failure for these components. Condition 1 also requires that the examination method to be used for these additional "Expansion" inspection category components be consistent with the examination method for the "Primary" inspection category component to which they are linked. The table below lists the CE and Westinghouse RVI components included under Condition 1. The staff requests that EPRI add the acceptance criteria, methodology, and data requirements for these components to Appendix C or Appendix E of the WCAP report as applicable, and add the corresponding flow charts to Appendix D and Appendix F of the WCAP report.

Component	**Link to "Primary" Inspection Category Components**
Upper core plate in Westinghouse-designed reactors	Control rod guide tube (CRGT) lower flange weld
Lower support forging or casting in Westinghouse-designed reactors	CRGT lower flange weld
Lower core support beams in CE-designed reactors	Upper core support barrel flange weld
Core support barrel assembly upper core barrel flange in CE-designed reactors	Upper core support barrel flange weld

RAI 46

Condition 2 of the draft SE for the MRP-227-Rev. 0 report requires that the components listed below to be moved from the "Expansion" inspection category to the "Primary" inspection category. Condition 2 also contains requirements for the examination methods, the examination coverage, and the examination frequency. The staff requests that EPRI modify the WCAP report to reflect this change.

Expansion Category Components Subject to IASCC and Neutron Embrittlement	Tables in the MRP-227-Rev. 0 report
Upper and lower core barrel welds in Westinghouse-designed reactors	Table 4-6
Lower core barrel flange weld in Westinghouse-designed reactors	Table 4-6
Core support barrel assembly lower cylinder welds and upper core barrel flange in CE-designed reactors	Table 4-5

RAI 47

Condition 3 of the SE of the MRP-227-Rev. 0 report required that certain high consequence of failure components subject to important combinations of multiple aging degradation mechanisms that were originally binned in the "Expansion" inspection category be included in the "Primary" inspection category. Condition 3 also contains requirements for the examination methods, the examination coverage, and the examination frequency. The following CE component is subject to Condition 3:

Component	Relevant Table and MRP Report
Core support column (casting or wrought) welds in lower support structure in CE-designed reactors	Table 4-5

The staff requests that EPRI modify the WCAP report to reflect this change.

RAI 48

- Appendix C Item CE-ID: 2 Core Shroud Assembly (Welded)
- Appendix C Item CE-ID: 2.1 Category: Core Shroud Assembly (Welded) – Remaining Axial Welds (pp. C-10 to C-11)
- Appendix C Item CE-ID: 3 Core Shroud Assembly (Welded) – Shroud Plates (pp. C-12 to C-13)
- Appendix C Item CE-ID: 3.1 Core Shroud Assembly (Welded) – Remaining Axial Welds, Ribs and Rings (pp. C-14 to C-15)

For the components listed above, the RVI geometry requires visual examination of these welds to be one-sided (from the shroud ID). Given that the examination is one-sided, without a stress analysis or supplementary UT examination, how can it be assured that cracks have not initiated on the opposite (OD) surface of the weld if the examination of the ID surface reveals no flaws or cracks?

RAI 49

- Appendix C Item CE-ID: 6.1 Core Support Barrel Assembly – Lower Core Barrel Flange (pp. C-22 to C-23)
- Appendix E Item W-ID: 3.1 Core Barrel Assembly – Core Barrel Flange, Core Barrel Outlet Nozzles, Lower Core Barrel Flange Weld (pp. E-12 to E-13)

For the items listed above, the corresponding flow charts in Appendix D, p. D-11 and Appendix F, p. F-8, show that crack-like indications are assumed to be through-wall. However, this is not clear in the written analysis procedures on pages C-22 and E-12.

The staff therefore requests that EPRI make the analysis procedures for the items listed above consistent with the flow charts for these items.