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.