ELECTRIC POWER RESEARCH INSTITUTE

MRP Materials Reliability Program_____MRP 2010-031
(via email)

April 20, 2010

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

PROJ 669

SUBJECT: EPRI MRP Final Draft Responses to: REQUEST FOR ADDITIONAL INFORMATION RE: ELECTRIC POWER RESEARCH INSTITUTE TOPICAL REPORT 1016596,'MATERIALS RELIABILITY PROGRAM: PRESSURIZED WATER REACTOR INTERNALS INSPECTION AND EVALUATION GUIDELINES (MRP-227-REV. 0)' (TAC NO. ME0680), November 12, 2009

To Whom It May Concern:

Enclosed are two copies of the subject document. This document is being provided to support MRP discussion with the NRC in a meeting scheduled for June 8-9, 2010.

If you have any questions on this item, please contact Christine King (cking@epri.com, 650-855-2164), or Chuck Welty (cwelty@epri.com, 650-855-2371).

Sincerely,

Terry McAllister
SCANA
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D035
NRR

Add'l copy sent to PM

Final Draft
Responses to 3rd Set RAIs
On MRP-227, Rev 0

April 21, 2010

MRP-227 RAI DRAFT Set #3 Responses
04/01/10

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

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

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

RAI 3-1 It is not evident to the NRC staff whether or not the inspection and evaluation (I&E) methodology in TR MRP-227-Rev. 0, or in other MRP report methodologies that support the TR MRP-227, Rev. 0, methodology, are:

> a) in compliance with the scoping, screening, and aging management requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 54, or;
> b) in conformance with the NRC staff's aging management recommendations in NUREG-1800, Revision 1, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," (SRP-LR), Section A.1, "Aging Management Review – Generic (Branch Technical Position (BTP) RLSB-1)."[1]

Perform a comparison of the recommended I&E methodology in TR MRP-227, Rev. 0, and its supporting MRP aging management methodologies to the NRC staff's recommended aging effect identification and management guidance in SRP-LR BTP RLSB-1. Demonstrate the compliance of the TR MRP-227, Revision 0, methodology with the requirements of 10 CFR Part 54 and with the NRC staff's recommendations in SRP-LR BTP RLSB-1. In addition, justify any non-compliances of your proposed methodology with the stated requirements in 10 CFR Part 54 or non-conformances with the recommendations in SRP-LR BTP RLSB-1.

Response: MRP-227, Revision 0 is intended to provide the technical basis for revisions to the NRC Standard Review Plan for License Renewal (NUREG-1800, Revision1). In particular, MRP-227, Revision 0 provides information required to augment and complete the generic reactor internals Aging Management Review (AMR) and Aging Management Plan outlined in the Generic Aging Lessons Learned (GALL) Report (NUREG-1801, Revision 1). NUREG-1800, Revision 1, and NUREG-1801, Revision 1 outline an acceptable process for implementing the aging management requirements of Title 10 of the *Code of Federal Regulations* Part 54 (10 CFR Part 54). In addition, the processes used in developing MRP-227, Revision 0 are in conformance with the recommendations in SRP-LR BTP RLSB-1.

It is important to note that NUREG-1801, Revision 1 specifically states that:

> *The GALL Report does not address scoping of structures and components for license renewal. Scoping is plant specific, and the results depend on the plant design and current licensing basis. The inclusion of a certain structure or component in the GALL Report does not mean that this particular structure or component is within the scope of license renewal for all plants. Conversely, the omission of a certain structure or component in the GALL Report does not mean that this particular structure or component is not within the scope of license renewal for any plants.*

The reactor internals components addressed in MRP-227 are defined in Section 2.3, which states that:

> *These guidelines are applicable to the reactor internal structural components; they do not address fuel assemblies, reactivity control assemblies, or welded attachments to the reactor vessel.*

[1] Henceforth, the reference to this Branch Position in this set of RAIs will be abbreviated as SRP-LR BTP RLSB-1.

In other words, no items are screened out of MRP-227, Revision 0, other than fuel assemblies, reactivity control assemblies, and welded attachments to the reactor vessel.

The scoping and screening requirements of 10 CFR Part 54 are specifically discussed in Section 2.1 of NUREG-1800, Revision 1. Section 3.1 of the same document recognizes the reactor internals AMR outlined in the current version of the GALL Report. This generic AMR was developed on the assumption the reactor internals assembly as a whole is considered to be a passive, safety related structure. Therefore, it is inferred that all reactor internals components are included under the scope of 10 CFR 54 and should be included within the AMR scope. Similar statements are included in aging management evaluations of reactor internals, such as WCAP-14577 Rev.1-A and BAW-2248A, which have been accepted as generic references for license renewal programs by the NRC. MRP-227, Revision 0 is consistent with this understanding. Complete lists of components included in the generic evaluations for B&W, CE and Westinghouse plants are provided in MRP-189, Revision 1 and MRP-191. However, MRP-227 does not address the scoping and screening requirements of 10CFR54 on an individual plant basis because such actions are part of a licensee's License Renewal Application.

Section 3.01 of the SRP-LR (NUREG-1800, Revision 1) defines the AMR process as follows:

> *This AMR consists of identifying the material, environment, aging effects, and the AMP(s) credited for managing the aging effects.*

As part of the effort to support the development of the MRP-227, Revision 0 guidelines, a comprehensive evaluation of material, environment and aging effects for PWR reactor internals components was completed, as represented by MRP-189, Revision 1 and MRP-191. This effort was similar to that included in previous generic AMRs. The results of those evaluations, which are reflected in the inspection recommendations outlined in the disposition tables provided in Section 3 of MRP-227, Revision 0 (Tables 3-1, 3-2 and 3-3) are consistent with the previous studies and the current GALL tables.

An AMP is required when an effect is identified that requires specific management activities. For components within the scope of the regulation, 10 CFR 54 Section 21 requires an Integrated Plant Assessment (IPA) to assure:

> *(3) For each structure and component identified in paragraph (a)(1) of this section, demonstrate that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation.*

The MRP-227, Revision 0 IPA methodology is based on a detailed evaluation to characterize the components within the structure subject to aging effects. This process implements the guidance given in SRP-LR BTP RLSB-1, Section A.1.2.1:

> *The determination of applicable aging effects is based on degradations that have occurred and those that potentially could cause structure and component degradation. The materials, environment, stresses, service conditions, operating experience, and other relevant information should be considered in identifying applicable aging effects. The effects of aging on the intended function(s) of structures and components should also be considered.*

Section 3.1.2.2 of the SRP-LR (AMR Results for Which Further Evaluation is Recommended by the GALL Report) contains five subsections with the following statement:

> *The GALL Report recommends no further aging management review if the applicant provides a commitment in the FSAR Supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.*

Those five subsections are:

3.1.2.2.6 *Loss of Fracture Toughness due to Neutron Irradiation Embrittlement and Void Swelling*

3.1.2.2.9 *Loss of Preload due to Stress Relaxation*

3.1.2.2.*12* *Cracking due to Stress Corrosion Cracking and Irradiation-Assisted Stress Corrosion Cracking (IASCC)*

3.1.2.2.15 *Changes in Dimensions due to Void Swelling*

3.1.2.2.17 *Cracking due to Stress Corrosion Cracking, Primary Water Stress Corrosion Cracking, and Irradiation-Assisted Stress Corrosion Cracking*

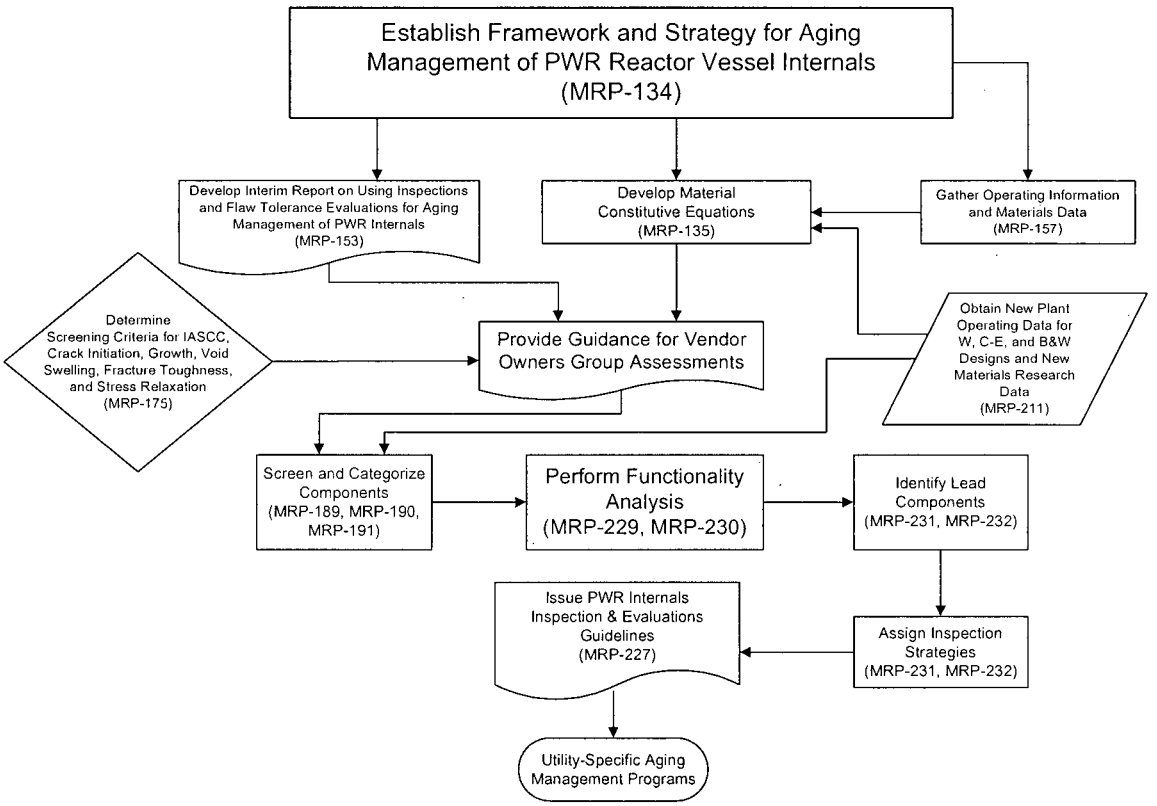
MRP-227, Revision 0 summarizes the results of industry programs for investigating and managing these aging effects in reactor internals and provides recommendations for the implementation of those results. A graphical representation of the process is depicted in MRP-227, Revision 0 Figure 2-1. The MRP-227, Revision 0 methodology begins with an extensive effort to identify materials, environment, stresses, service conditions, operating experience and other relevant information as required by SRP-LR BTP RLSB-1. This initial effort is described in MRP-134, MRP-135, MRP-153, MRP-157 and MRP-211. The second step in the process defines screening criteria that can be used to identify conditions that can potentially result in component degradation. Screening criteria were developed for eight relevant degradation mechanisms. These relevant degradation modes included the mechanisms identified in the GALL as "AMR Results for Which Further Evaluation is Recommended." Each component was compared to the MRP-175 screening criteria to identify applicable aging effects. The results of the screening process are reported in MRP-189, Revision 1 and MRP-191. This screening process is integral to the AMR and should not be confused with the screening process used to identify passive components per 10 CFR Part 54 21(a)(1).

Also in accordance with SRP-LR BTP RLSB-1, the effects of aging on the intended function(s) of the structure and components was evaluated using a FMECA (Failure Modes, Effects and Criticality Analysis) process. This FMECA process adds a level of further review that is extremely useful in prioritizing components and component

groupings prior to aging management program definition. SRP-LB BTP RLSB-1 specifically states that: "The risk significance of a structure or component could be considered in evaluating the robustness of an aging management program."

The transition from "applicable aging management effects" represented by MRP-189, Revision 1, MRP-190, and MRP-191 to "aging management program elements" is provided by MRP-231 and MRP-232, from which the requirements of MRP-227, Revision 0 were extracted and refined. The inspection guidelines provided in Section 4 of MRP-227, Revision 0 provide the basis for the program elements required in an AMP. Appendix A of MRP-227, Revision 0 summarizes the AMP Program Attributes required by NUREG-1800, Revision 1. The MRP-227, Revision 0 document is intended to provide the technical basis for an AMP. This appendix describes the extent to which the MRP-227 guidelines and supporting documents satisfy the AMP requirements. The MRP has agreed to work with the NRC to have these AMP recommendations incorporated in the upcoming GALL revision.

Figure 2.1 from MRP-227



RAI 3-2 TR MRP-227, Rev. 0, does not make reference to Nuclear Energy Institute (NEI) Report No. NEI 95-10, Revision 6,[2] 2 as an applicable industry license renewal report; nor does TR MRP-227, Rev. 0, provide any discussion on how it will be used to conform to the recommendations in NEI 95-10, Rev. 6, when implementation of the I&E recommendations are applied as license renewal activities and methods for aging management of RVI components Discuss how and provide the basis for why the recommended I&E scoping/screening process, inspection activities, evaluation methods and acceptance criteria in TR MRP-227, Rev. 0, are considered to be in conformance with the recommendations in NEI 95-10, Rev. 6, paying particular attention to conformance with Chapters 2, 3, and 4 of the NEI 95-10, Rev. 6, and the license renewal processes identified therein for implementation in Figures 2.0-1, 3.0-1, 4.1-1, 4.2-1, and 4.3-2.

Response: NEI 95-10, Revision 6, provides useful information and guidance to license renewal applicants on the steps to be followed in order to meet 10 CFR Part 54 requirements. The document was used by the team that developed MRP-227 and the supporting documentation. The basic process is outlined in the identified implementation figures.

The overall license renewal process is outlined in Figure 2.0-1. There are two major branches in Figure 2.0-1. The first branch describes aging management activities including the development of AMRs and associated AMPs. The second branch describes the process used for Time Limited Aging Analysis (TLAA). MRP-227, Revision 0 and its supporting documents provide the technical basis for aging management of systems structures and components within the reactor internals. This objective of the MRP effort was to provide documents that would support utilities in the implementation process and allow appropriate revisions to the NUREG-1801, Revision 1 (GALL) report. While MRP-227, Revision 0 is not intended to follow the implementation process precisely, it is certainly supports the aging management branch of the license renewal process. Its recommendations are consistent with the intent of the process and assure utility conformance in the license renewal process.

The method used to identify systems, structures and components within the scope of the license renewal rule is described in Section 3 of NEI 95-10 and outlined in Figure 3.0-1. Note that the process described in Figure 3.0-1 does not distinguish between systems, structures and components. The footnote to the rule in the Federal Register contains the following clarification:

> *The Commission intends that the phrase, ''systems, structures, and components'' applies to the matters involving the discussions of the overall renewal review, the specific license renewal scope (§ 54.4), time-limited aging analyses (§ 54.21(c)), and the license renewal finding (§ 54.29). The phrase, ''structures and components'' applies to matters involving the integrated plant assessment (IPA) required by § 54.21(a) because the aging management review required within the IPA should be a component and structure level review rather than a more general system level review.*

[2] NEI 95-10, Revision 6, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54-The License Renewal Rule." NEI 95-10, Rev. 6, has been endorsed for use in Regulatory Guide 1.188, "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses"[Sept. 2005].

The intention of this section is clearly to provide flexibility in the determination of systems, structures and components considered in the scope of the rule. As indicated in the response to RAI 3-1, the scope statement of MRP-227 includes the entire reactor internals structure. The function of the reactor internals structure is defined in MRP Section 3.1:

The functions of PWR internals are to:

1. *provide support, guidance, and protection for the reactor core;*
2. *provide a passageway for the distribution of the reactor coolant flow to the reactor core;*
3. *provide a passageway for support, guidance, and protection for control elements and in vessel/core instrumentation; and*
4. *provide gamma and neutron shielding for the reactor vessel.*

As these functions are critical to the operation of the reactor, the reactor internals assembly as a whole is considered safety related. Therefore, the reactor internals are within the scope of 10 CFR 54.4(a). The functions of the system have been identified per the requirements of 10 CFR 54.4(b). More detailed considerations of component function come under Section 4 of NEI 95-10.

Figure 4.1-1 describes the NEI 95-10 process for identifying specific structures and components and their function. Both the SRP-LR (NUREG-1800, Revision 1) and the GALL Report (NUREG-1801, Revision 1) identify the reactor internals as part of the Reactor Vessel, Internals and Reactor Coolant system. The reactor internals include several recognized structures (e.g., core barrel assembly, control rod guide tubes). However, the MRP analysis was completed on a component by component basis. The functions of the components and their relevance to the function of the reactor internals as a whole are described in MRP-189, Revision 1 and MRP-191. Table 2.1-5 of NUREG-1800, Revision 1 identifies the reactor internals as a "Structure, Component or Commodity that meets the Requirements of 10CFR54.21(a)(1)(i)". The reactor internals components are all passive and are not replaced on a regular basis and are, therefore, by the criteria of 10CFR54.21(a)(1)(i) and (ii), subject to management review. The NEI 95-10 process suggests that commodity groupings may be applied to the items requiring management review. The structure of the MRP review does contain several examples of large groupings of components. For instance the baffle bolting is listed as a single component item. In practice there may be hundreds of baffle bolts and there may be large variations in the stresses and irradiation exposure of the individual bolts. Similar variations may exist for reactor internals structures such as the lower support columns or the control rod guide tube assemblies. The structure of the analysis should be evident in MRP-189, Revision 1 and MRP-191.

A large portion of the MRP effort has been devoted to identification of those aging effects that require management. The MRP process was consistent with the upper portion of Figure 4.2-1 from NEI 95-10. The screening and FMECA processes described in the response to RAI 3-1 are consistent with the NEI 95-10 process. Tables 3.1, 3.2 and 3.3 of MRP-227 identify the relevant aging effects for the reactor internals.

The process used by the utility to evaluate available AMPs is described n Figure 4.3-2. The intent of the MRP effort was to define AMP options for the anticipated revision of NUREG-1801, Revision 1. Section 4.3.2 outlines the general requirements for a Plant

Specific Aging Management Program. The inspection recommendations outlined in MRP-227, Revision 0 meet these general requirements. The bases for the inspection strategies are outlined in MRP-231 and MRP-232.

Figure 2.0-1
LICENSE RENEWAL IMPLEMENTATION PROCESS

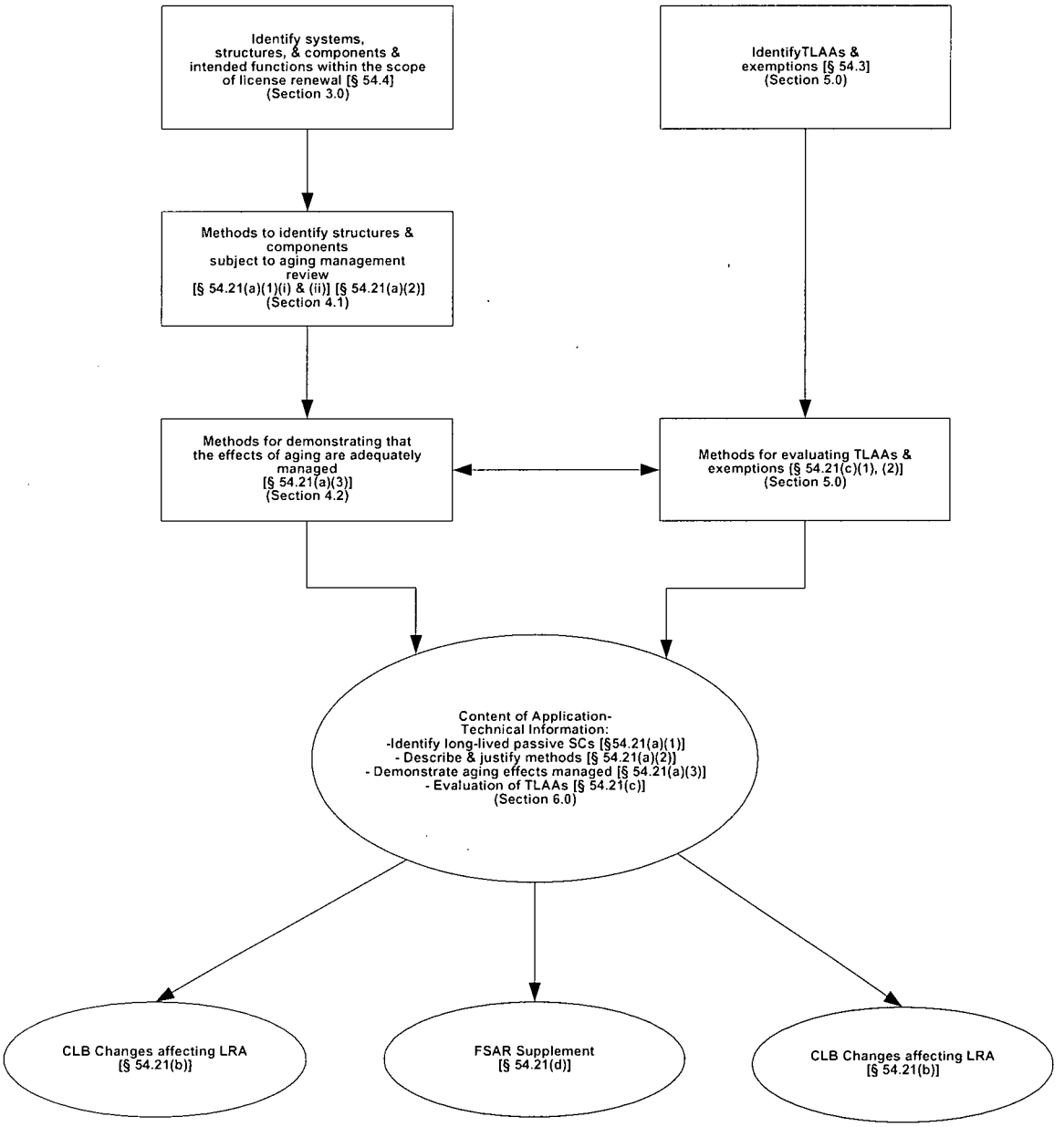


Figure 3.0-1
A METHOD TO IDENTIFY SSCs AND INTENDED FUNCTIONS WITHIN THE SCOPE OF LICENSE RENEWAL [10 CFR 54.4(a) & (b)]

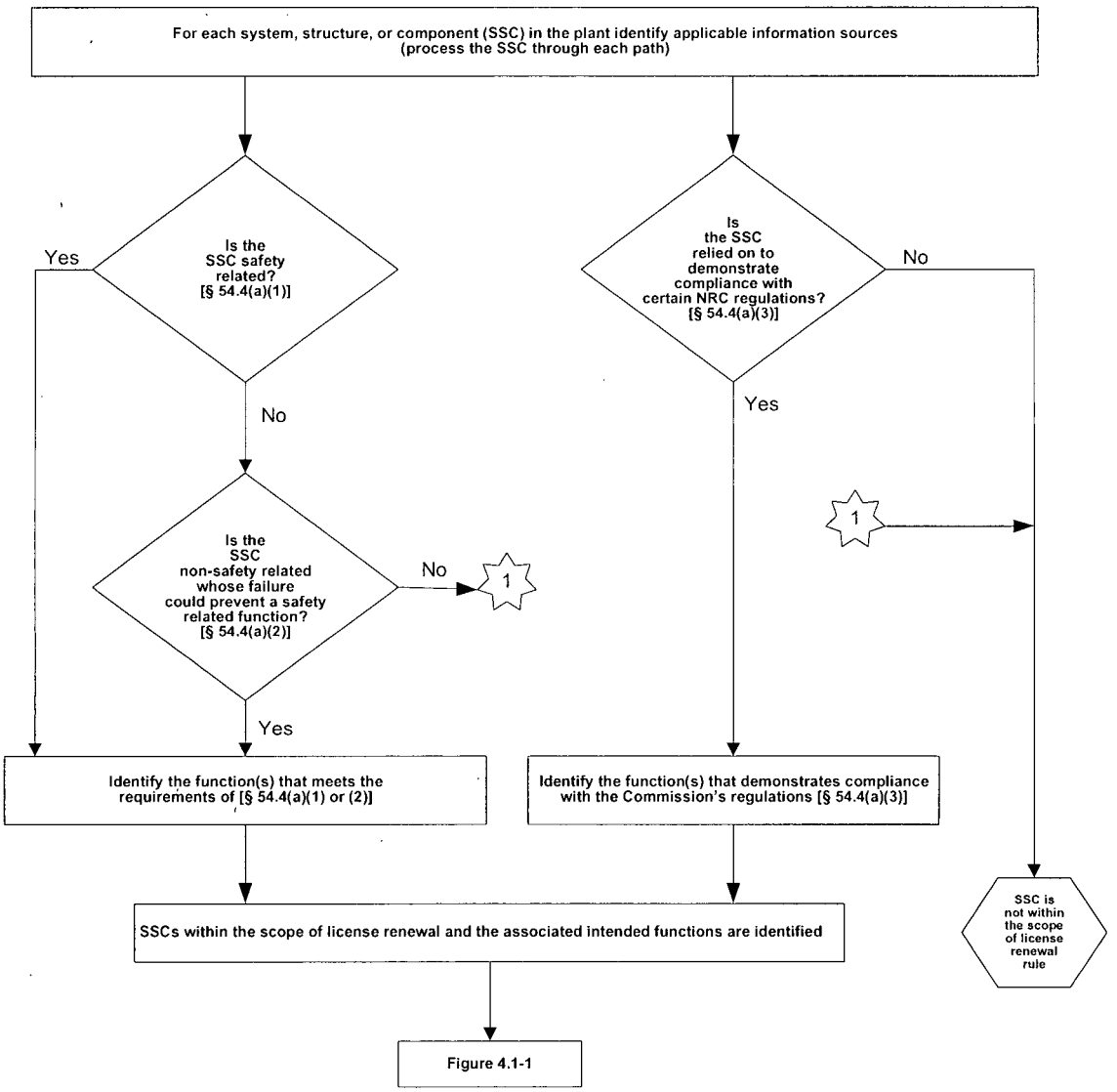


Figure 4.1-1
A METHOD TO IDENTIFY SSCs AND INTENDED FUNCTIONS WITHIN THE SCOPE OF LICENSE RENEWAL [10 CFR 54.4(a) & (b)]

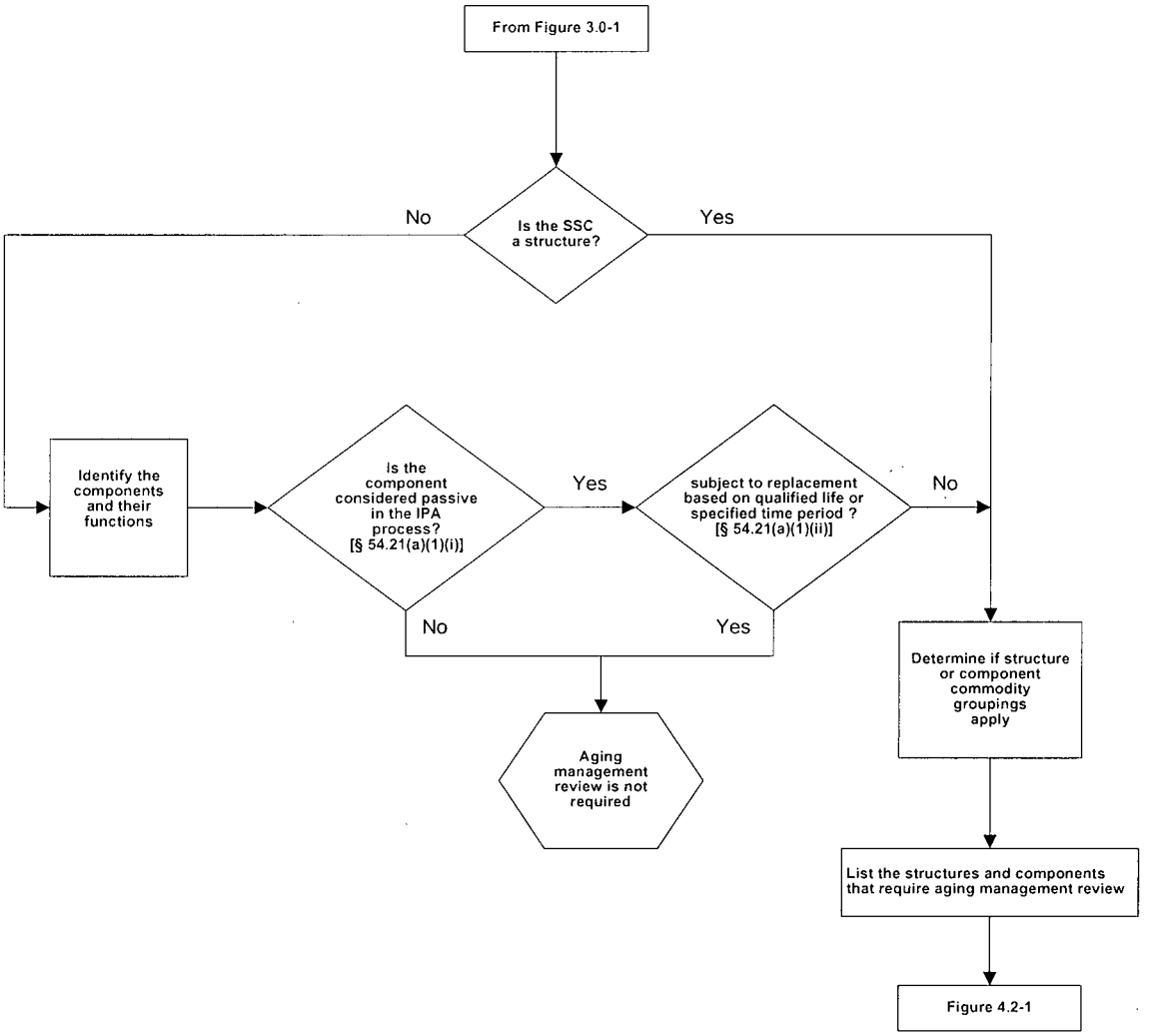


Figure 4.2-1
Identification of Aging Effects Requiring Management

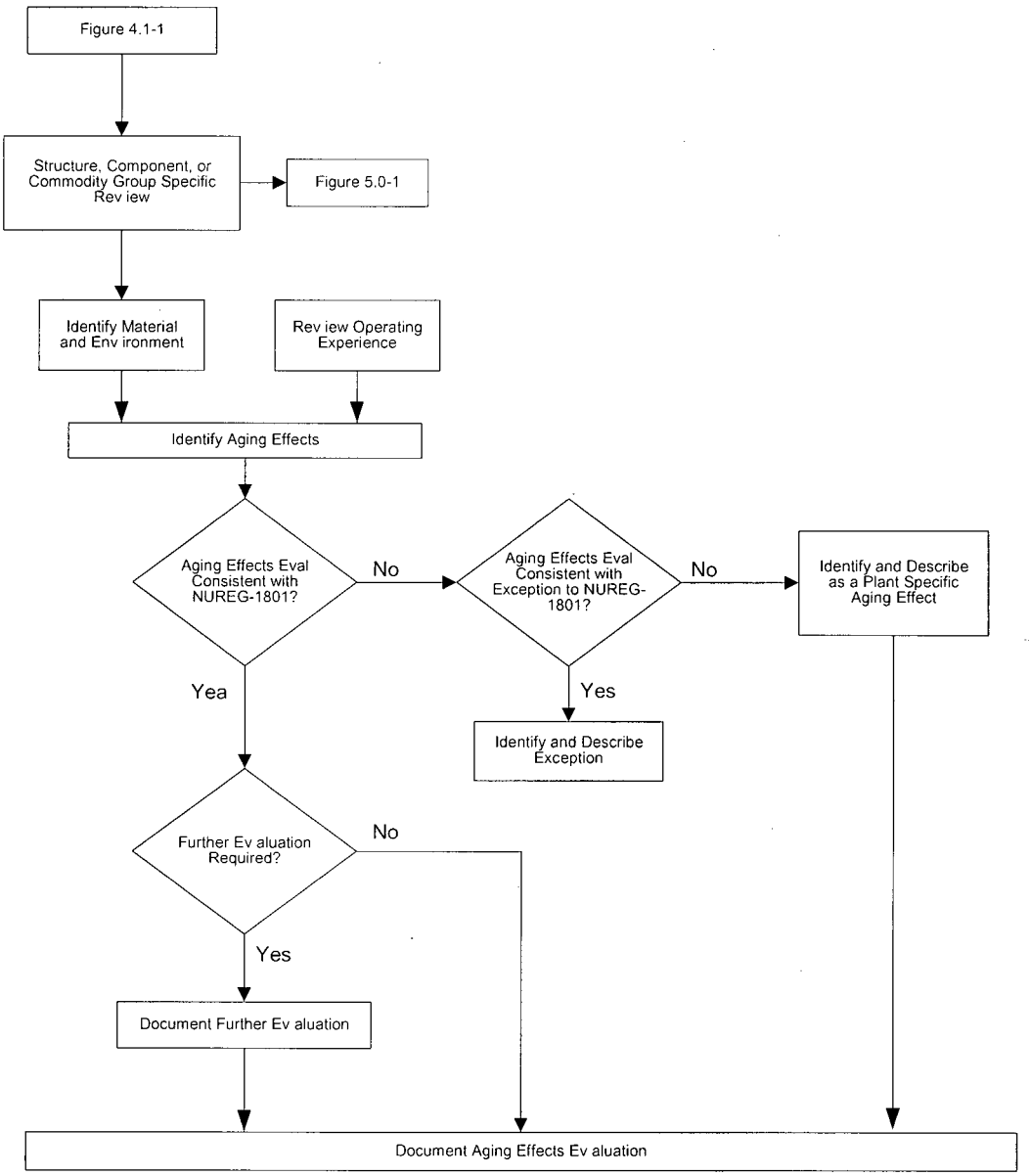
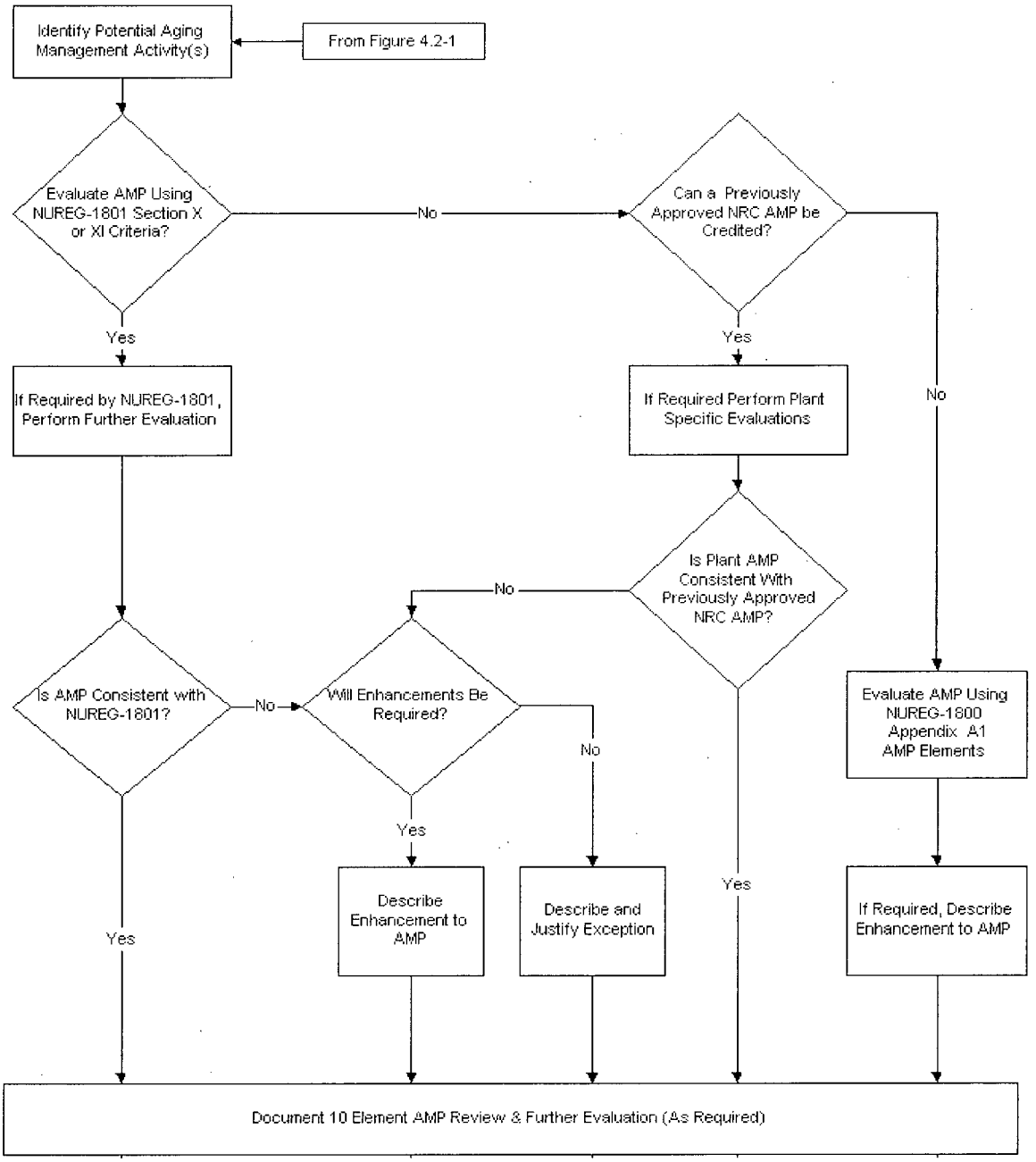


FIGURE 4.3-2
Aging Management Program Review



RAI 3-3 By letter dated February 20, 2009, the NRC granted a fee waiver under the provisions of 10 CFR 170.11(a)(1)(iii) to support a non-fee billable review of TR MRP-227, Rev. 0. The fee waiver was granted by the NRC on the basis that the methodology of the report would be used to update the NRC staff's aging management review (AMR) items for PWR RVI components, as given in the following NRC license renewal guidance documents: (1) the SRP-LR; and (2) NUREG-1801, "Generic Aging Lessons Learned Report" (GALL), Volumes 1 and 2.

The NRC's recommended AMR items in Tables of the GALL Report, Revision 1, Volume 1 (henceforth referred to as Table 1 AMR items) are given in the following AMR column format:

Table X. Summary of Aging Management Programs for the . . . . System Evaluated in Chapter IV of the GALL Report							
ID	Type	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Required	Related Generic Item	Unique Item

The NRC's recommended AMR items in Tables of the GALL Report, Revision 1, Volume 2 (henceforth referred to as Table 2 AMR items) are given in the following AMR column format:

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM B2 (or B3; B4) Reactor Vessel Internals (PWR) – Westinghouse (Combustion Engineering; Babcock and Wilcox)							
Item	Link	Structure and/or Component	Material	Environment	Aging Effect/Mechanism Required	Aging Management Program (AMP)	Further Evaluation

These AMR item formats have been adopted for use in Tables 3.0-1 and 3.0-2 of NEI 95-10, Revision 6, and endorsed for use in Regulatory Guide 1.188, Revision 1.

Provide Table 1 AMR items for Westinghouse (W), Combustion Engineering (CE), and Babcock & Wilcox (B&W) RVI component commodity groups that are in conformance with the guidelines for formatting Table 1 AMR items in Figure 3.0-1 of NEI 95-10, Revision 6. The NRC staff also requests that you provide the Table 2 AMR items for W, CE, and B&W RVI component commodity groups that are in conformance with the guidelines for formatting Table 1 AMR items in Figure 3.0-2 of NEI 95-10, Revision 6.

Response: MRP Letter MRP-2009-091, Dennis Weakland (MRP) to Tanya Mensah (NRC), dated December 2, 2009, quoted in part below, provided material that is responsive to this RAI:

"The following five documents provide the MRP's initial draft input to assist the NRC staff in updating NUREG 1801, "Generic Aging Lessons Learned Report" (GALL):

1) EPRI DRAFT Input (12-01-09): GALL Chapter XI.M16
2) EPRI DRAFT Input (12-01-09): GALL Table IV.B2 (W)
3) EPRI DRAFT Input (12-01-09): GALL Table IV.B3 (CE)
4) EPRI DRAFT Input (12-01-09): GALL Table IV.B4 (B&W)
5) EPRI DRAFT Input (12-01-09): New Appendix A to MRP-227

These documents have been forwarded to the Document Control Desk by the referenced letter (copy attached).

RAI 3-4 Provide a new draft GALL AMP XI.M16, "PWR Vessel Internals Program" in a format that conforms to the recommended program element criteria in SRP-LR BTP RLSB Section A.1.2.3 and that can be adopted for the contents of an applicant's PWR Vessel Internals Program when the license renewal application is submitted to the NRC for NRC staff approval.

Response: See response to RAI 3-3 above.

RAI 3-5 Provide the basis for why the functionality analysis (FA) in Figure 2-2 was applied to Category C RVI components and not to Category B RVI components.

Response: The process outlined in Figure 2-2 was used to categorize components based on the "likelihood and severity of safety and economic consequences". Priority in the functionality analysis was given to the Category C components because they were found to be "potentially significantly affected". However, the analysis was not limited to the Category C items. A lower priority was given to Category B components because the consequences were found to be "potentially moderately significant". SRP-LB BTP RLSB-1 specifically states that: "The risk significance of a structure or component could be considered in evaluating the robustness of an aging management program."
The detailed functionality analysis of the barrel former and core shroud incorporated both Category C and Category B components. In other cases, it was determined that existing analysis of Category B and C components was sufficient and additional functionality analysis was not required. For instance, although no detailed mechanical analysis of the lower support columns was undertaken in this study, existing analysis indicates that the stresses in these components are primarily compressive.

Because the initial screening and categorization process identified some level of potential safety and economic consequence for all of the Category B and C components, these components were all considered in the development of the MRP-227 program. The disposition of all Category B and C components is shown in Tables 3-1, 3-2 and 3-3. The final disposition was based on the results of the functionality analysis combined with component accessibility, operating experience, existing evaluations and prior examination results.

RAI 3-6 Clarify whether or not the existing methodology in TR MRP-227, Rev. 0, can be applied to a PWR facility whose reactor core loading pattern operating history is not bounded by the assumptions in the report. If the methodology can be applied, justify why that is the case. If the methodology cannot be applied to these PWRs, identify what actions a licensee with a non-conforming PWR would have to take in order to develop a plant-specific AMP for its RVI components, which is consistent with the intent of TR MRP-227, Rev. 0. Identify whether license renewal applicants should demonstrate that their facility's reactor core loading pattern operating history is bounded by the assumptions in the report as part of the license renewal application (i.e., should be a license renewal applicant action item).

Response: Similar concerns were expressed in RAI 2-19 from the 8/24/09 inquiries. Based on the previous discussion the following conclusions may be drawn:

1. Section 2.4 of MRP-227 states: "The recommendations are thus applicable to all U.S. PWR operating plants as of May 2007 for the three designs identified."
2. Section 2.4 further states: "Plant modifications made or considered after this date should be reviewed to assess impacts on strategies contained in these guidelines."
3. The strategies in MRP-227 do not assume that the core loading patterns used in the analysis are bounding.
4. The inspection recommendations are robust and there is no reason to anticipate plant modifications that would impact the MRP-227 requirements.
5. To apply MRP-227, the license renewal applicant needs to demonstrate that core loading patterns going forward are reasonably represented by the assumptions of the report.
6. MRP-227 is a living document and the industry will monitor any trends in operating practice that might impact the MRP-227 recommendations.

Original Response to 8/24/09 RAI 2-19:
The core loading patterns used in the MRP-227 reference documents were chosen to represent known operating practice, they are not intended to be used as a reference for plant-specific analysis. The intention of using the representative core loading patterns was not to bracket operation, but to perform an analysis that demonstrates both historic and current fuel management programs. The MRP-227 inspection recommendations based on these calculations are robust and do not require the utility to perform additional analysis of core loading patterns to qualify their applicability.

The condition of the internals at the time of the first required inspections is dominated by the power distribution used to represent the first thirty years of full power operation. During this period the analysis assumed that the fresh fuel was loaded in the peripheral fuel assemblies. This "out-in" loading pattern produced results in relatively high heat loadings and neutron fluences in the near core structure. In practice all plants in the United States abandoned fuel management based on the "out-in" loading prior to thirty years of operation. There are no current or planned fuel management programs that would result in more deleterious conditions than those assumed in this analysis during the first thirty years of operation. For this reason there is no reason to require any plant to perform an analysis to demonstrate adherence to the assumed core loading pattern prior to performing the first round of inspections. The timing and extent of the first round of MRP-227 examinations is governed by damage that has already been accumulated.

The representative power distributions used for the simulation of years 31 to 60 incorporate the effects of aggressive power uprate programs. Qualification of the core loading pattern is considered in the design analysis for the plant uprate. Although it is not possible to anticipate all possible future options, both current fuel management practice, which maximizes fuel utilization, and concerns about neutron damage in the reactor pressure vessel preclude return to the practice of loading fresh fuel in the periphery locations. It is unlikely that future core loading patterns would invalidate the assumptions of the analysis.

Although the shift from "out-in" core loading patterns to low-leakage patterns resulted in a sharp decrease in the peak temperature in the internals structure, the shift had minimal effect on the location of the peak temperature or the character of the peak damage. There

is no reason to expect that changing the loading pattern would change the base inspection recommendations. The MRP-227 recommendations are based on reasonable assumptions about the effects of power uprates. In many cases power uprates can be accomplished without significantly increasing the heat or neutron loading to the internals. Return to the more aggressive core loading patterns could conceivably result in a decrease in the re-inspection interval. However, there is no reason to anticipate any change of this scale.

MRP-227 is intended to be a living document. The MRP will monitor both inspection results and plant operating experience and make appropriate modifications. There is currently no need to require plants to demonstrate adherence to any reference core loading practice.

RAI 3-7 Alloy 600 PWR RVI components and their associated welds manufactured from Alloys 82 and 182 are susceptible to primary water stress corrosion cracking (PWSCC) when exposed to PWR reactor coolant water. In Table 3-1 of TR MRP-227, the following Babcock and Wilcox (B&W) Alloy X-750 PWR RVI components were welded with Alloy 82 material and yet they were classified under "N" category which excludes inspections for these PWR RVI components: (1) dowel-to-core barrel cylinder welds, (2) dowel-to-upper grid rib section bottom flange welds, (3) dowel locking welds, (4) dowel-to-guide block welds, and (5) dowel-to-distributor flange welds. Even though stress levels in these components may not exceed the threshold levels, the NRC staff considers it to be likely that PWSCC can potentially occur due to the introduction of cold work during fabrication. In light of this observation, provide an explanation for excluding inspection requirements for these B&W PWR RVI components.

Response: The explanation for categorization of these dowel welds is provided in MRP-231-Rev. 1. Excerpts are provided below:

Many preliminary Non-Category A welds in Table 1-2 used nickel-based Alloy 69 (INCO 69) and Alloy 82 (INCO 82) materials, which are susceptible to PWSCC (or SCC as listed in Table 1-2). However, some of these Alloy 69 and Alloy 82 welds are for locking Alloy X-750 alignment dowels which facilitated the internals assembly process. These dowels do not have any function after the internals items were joined by bolting. Hence, those preliminary Category B welds were changed to Category A. The updated Non-Category A welds are included in Table 2-8 with the following welds no longer listed. It should be noted that the reclassified Category A welds below based on the functionality assessment are called "No Additional Measures (N)" in Section 3 and are listed in Table 3-8 as "N".

Plenum Cover Assembly

Alloy X-750 dowels-to-plenum cover bottom flange welds

These welds are used for locking two Alloy X-750 dowels, which were used to align the plenum cover bottom flange with the plenum cylinder top flange. After the plenum cover bottom flange is bolted to the plenum cylinder top flange with 64 bolts, the two Alloy X-750 dowels and their locking welds no longer have any function.

Upper Grid Assembly

Alloy X-750 dowel-to-upper grid rib section bottom flange welds

These welds are used for locking two Alloy X-750 dowels, which were used to align the upper grid rib section with the upper grid ring forging. After the upper grid rib section is bolted to the upper grid ring forging with 36 cap screws, the two Alloy X-750 dowels and their locking welds no longer have any function.

Core Barrel Assembly

Alloy X-750 core barrel-to-former plate dowels and the locking welds

These welds are used for locking the 32 Alloy X-750 dowels, which were used to align the former plates with the core barrel cylinder at the top and bottom former plate level (16 dowels at each level). After the former plates are bolted to the core barrel cylinder with the CB bolts, these Alloy X-750 dowels and their locking welds no longer have any function. These dowels are not considered in the core barrel assembly functionality analysis.

Lower Grid Assembly

Alloy X-750 dowel-to-lower grid shell forging welds

These welds are used for locking two Alloy X-750 dowels, which were used to align the lower grid shell forging with the core barrel cylinder bottom flange. After the lower grid shell forging is bolted to the core barrel cylinder bottom flange with 108 LCB bolts, the two Alloy X-750 dowels and their locking welds no longer have any function.

Alloy X-750 dowel-to-lower grid rib section welds

These welds are used for locking two Alloy X-750 dowels, which were used to align the lower grid rib section with the lower grid shell forging. After the lower grid rib section is bolted to the lower grid shell forging with 36 cap screws, the two Alloy X-750 dowels and their locking welds no longer have any function.

Flow Distributor Assembly

Alloy X-750 dowel-to-flow distributor flange welds

These welds are used for locking two Alloy X-750 dowels, which were used to align the flow distributor flange with the lower grid shell forging. After the flow distributor flange is bolted to the lower grid shell forging with 96 LCB bolts, the Alloy X-750 dowels and the locking welds no longer have any function.

Table 3-7 in MRP-231 (Rev. 1) summarizes the remaining dowel welds, which remain as either Primary or Expansion items:

Table RAI 3-7-1
Summary of Nickel-Based Alloy Welds for PWSCC

Item	Table 2-8 Category	Final Category
Lower Grid Assembly Alloy X-750 Dowel to Guide Block Welds	B	P
Upper Grid Fuel Assembly Support Pads: Alloy X-750 Dowel Locking Weld (except DB)	B	E
Lower Grid Fuel Assembly Support Pads: Alloy X-750 Dowel Locking Weld (Note 1)	B	E

Note 1, Alloy X-750 Dowel Locking Weld is also listed in Section 3.2.6 for Irradiation Embrittlement, and is also categorized as "Expansion".

The lower grid assembly dowel-to-guide block locking welds serve as loose part prevention devices. They are not structural. The Alloy 82 locking welds may be susceptible to cracking as a result of stress corrosion cracking (i.e., PWSCC). Small cracks in the locking weld are acceptable since the locking function can be maintained as long as any part of the weld is present. The fillet welds are therefore categorized as Primary items.

The upper and lower grid assemblies Alloy X-750 dowel-to-fuel assembly support pad welds (either Alloy 82 or Alloy 69 material) are also considered loose part prevention devices and are categorized as Expansion items. Evidence of cracking is expected to occur sporadically and not lead to a short-term serious degradation. However, evidence of cracking should lead to enhanced vigilance for possible loose parts monitoring and consideration for implementing the repair scenarios.

RAI 3-8 In Section 2.4 of TR MRP-227, Rev. 0, the MRP assumes that the design of a PWR plant applying the TR MRP-227, Rev. 0, methodology would not include any design changes beyond those identified in either general industry guidance or recommended by the original vendors. The NRC staff is aware that many of the licensees owning PWR facilities have been granted license amendments to implement measurement uncertainty recapture (MUR) power uprates, stretch power uprates, or extended power uprates for their facilities. However, it is not evident to the NRC staff whether any design changes associated with these type of power uprates would be within the scope of the MRP's term "design changes identified in general industry guidance or recommended by original vendors." Clarify whether design changes that will need to be implemented in order to receive NRC approval of a MUR, stretch, or extended power uprate, or that have been implemented as a result of receiving NRC approval of a power uprate, are within the scope this type of assumption.

Response: The TR MRP-227, Rev. 0 recommendations were based on evaluations relevant to current plant operating experience at the time the report was issued. This operating experience includes measurement uncertainty uprates, stretch power uprates and extended power uprates. Therefore, all three uprate types are considered within the scope of "design changes identified in general industry guidance or recommended by original vendors." As it is not possible to anticipate all possible future plant uprates or

modifications, MRP-227 clearly states that "Plant-specific commitments remain the responsibility of the owner."

Average core power must be increased to implement a plant uprate. This increase in power necessarily implies an increase in the average neutron flux in the core. However, for those reactor internals components that are subject to neutron radiation damage, the neutron exposure tends to be determined by power levels in the peripheral fuel assemblies, rather than the core average power. The original "out-in" core loading patterns used in many plants produced relatively flat core power distributions. These flat power distributions lead to higher neutron leakage at the edges of the core. In addition to causing high damage rates in the internals, this leakage results in increased neutron exposure of the reactor pressure vessel and relatively poor fuel utilization. Current core design practice utilizes a "low leakage" loading that tends to reduce power levels in the peripheral assemblies, which in turn reduces the neutron exposure of the reactor internals. While plant uprates may lead to some increase in neutron exposure of the internals, these are increases of an exposure level already reduced by the core loading, and they are generally moderate compared to the overall increase in power level.

Measurement uncertainty recapture uprates take advantage of improved power monitoring systems to allow the plant to increase power inputs by operating closer to the plant allowables. As these uprates remain within the original design basis of the plant, there is no reason to believe that a measurement uncertainty recapture uprate would fall outside the scope of the MRP-227 recommendations.

Stretch power uprates take advantage of excess margins that are buried in the plant operating limits. In many cases, a plants operating below the true plant capacity can demonstrate safe operation at higher power outputs by removing overly conservative limits. In most cases, the stretch power uprates result in power increases in an individual plant, but do not move the plant outside the envelope of fleet operating experience. Therefore, there is no reason to believe that a stretch power uprate would fall outside the scope of the MRP-227 recommendations.

Extended power uprates produce the largest increases in plant power. An extended power uprate may rely on both more detailed analysis of plant operation and upgrades to plant equipment. The experience base considered in support of the MRP-227 recommendations included plants with extended power uprates. The core power distributions used in the modeling of irradiation-induced aging of the core baffle and shroud structures included a typical extended power uprate. It is the responsibility of the plant owner to demonstrate that the changes in plant operation are consistent with the general assumptions of MRP-227.

The finite element-based aging analysis of the core baffle-formers and core shroud completed in support of the MRP-227 guidelines were never intended to provide bounding plant results. The recommendations are robust and not dependent on the details of the analysis. The assumption that the representative plant operated for thirty years with "out-in" core loading patterns has a large impact on the results. Most plants moved away from this aggressive core loading pattern much earlier in plant life. The effects of this conservative assumption about plant operating history are generally larger than any potential effect of plant uprate.

RAI 3-9 The middle of page 4-1 of Section 4 of TR MRP-227, Rev. 0, provides a list of program element activities for the I&E methodology. SRP-LR BTP RLSB-1 Section A.1.2.3, *Program Elements,* provides the recommended program element criteria for AMPs. For each bulleted program criterion that is provided on page 4-1 of TR MRP-227, Rev. 0, the NRC staff requests that a reference or link be provided that matches the programmatic criterion in the bulleted list to its corresponding program element subsection in SRP-LR BTP RLSB-1 Section A.1.2.3. In addition, the NRC staff requests the criteria in the bulleted list be amended to include an aging management criterion that corresponds to the recommended program element criteria for condition monitoring programs in SRP-LR BTP RLSB-1, Section A.1.2.3.3, "*Parameters Monitored/Inspected.*"

Response: The inspection and evaluation activities shown as bullets in the middle of page 4-1 of Section 4 of MRP-227 were not intended to match completely with the aging management program elements of SRP-LR BTP RLSB-1, Section A.1.2.3, *Program Elements*. These bulleted items were intended only to characterize the "typical" concerns that should be addressed when developing an adequate aging management program. Instead, the industry has prepared a draft of Aging Management Program XI.M16 that describes an explicit comparison between the contents of MRP-227 and the *Program Elements*. This draft should be the point of comparison.

However, many of the typical aging management program concerns on page 4-1 can be compared to, or placed in the context of, SRP-LR aging management program attributes. For example, "selection of items for aging management" is similar to *Scope, although* additional detail on "selection of items for aging management" could eventually lead to *Parameters Monitored/Inspected*. The concern about "selection of the type of examination or other methodologies appropriate for each applicable degradation mechanism" is similar to *Detection of Aging Effects,* as are "specification of the required level of examination qualification" and "schedule of first and frequency of any subsequent examinations." The concern about "sampling and coverage" could be placed with *Parameters Monitored/Inspected* or with *Detection of Aging Effects*, depending on emphasis. The concern about "expansion of scope if sufficient evidence of degradation is observed" is probably best placed with *Corrective Actions*, while "examination acceptance criteria" belongs with *Acceptance Criteria* and "methods for evaluating examination results not meeting the examination acceptance criteria" also is best placed with *Corrective Actions*. The concern about "updating the program based on industry-wide results" matches well with *Operating Experience*, while "contingency measures to repair, replace, or mitigate" also fits with *Corrective Actions*. It should be repeated that the list on page 4-1 was not intended to parallel in any way the SRP-LR aging management program attributes.

RAI 3-10 Section 4.1.3, *Aging Management Methodology Qualification*, of TR MRP-227, Rev. 0, (page 4-2) provides the inspection method qualification discussion and basis for the report. The section implies that methodologies may be qualified in accordance with appropriate qualification requirements, standards, or procedures (e.g., the qualification requirements in the ASME Code), or may be qualified by only the development of a technical justification to explain the applicability of the selected methodology. Please amend the stated section to be more specific on the qualification methods that would be used to qualify a given inspection technique for implementation for both the case where an RVI component is scoped in for license renewal under one of the safety related intended functions mentioned in either 10 CFR 54.4(a)(1)(i),(ii), or (iii),

and for the case where a non-safety related RVI component is scoped in for license renewal under the scoping requirement of 10 CFR 54.4(a)(2).

Response: The intent of Section 4.1.3 is to demonstrate that the inspection methodologies specified in MRP-227 for aging management are methodologies that are commonly used in industry and have a substantial experience base.

MRP-227 identifies the aging effects to be detected while its companion document, MRP-228, identifies the experience base for each of the examination methodologies selected, determines the level of standardization and contains the requirements specific to the inspection methodologies involved, as well as requirements for qualification of the NDE systems used to perform those inspections. Whereas failure of component items individually often do not affect the ability of the component/assembly to perform its function, there is no differentiation applied between safety related or non-safety related internals examination requirements in either MRP-227 or MRP-228. All components currently selected for examination in MRP-227 are either safety-related or important to safety. For the purposes of specifying qualification of examinations, reactor internals inspections (PWR or BWR) are divided into either; "remote visual" examinations or "non-remote visual" examinations.

For remote examination procedures the generic requirements for visual examination (EVT-1, VT-1, and VT-3) described in Section 2.3 shall be met, including those addressing the personnel training and experience requirements for individuals performing those examinations.

For examinations other than remote visual, a Technical Justification is required for each examination procedure in accordance with Section 2.1.

No inspections have been specified by MRP-227 that would require a higher level of qualification than a technical justification. The use of ASME Section V Article 14 in conjunction with MRP-228 provides a standardized process whereby components selected for examination by means other than visual must use the technical justification process. Utilities and/or vendors can include more rigor in order to achieve successful implementation of the examination requirements, but not less.

The use of ASME Section V Article 14 in conjunction with MRP-228 provides a standard process for qualification of known methodologies (Technical Justifications) but also provides for scenarios where a component may be examined using an unforeseen new methodology, an untried delivery device, or a more discerning supplemental or alternate examination without a substantial experience base. In such cases the utility or vendor could specify a higher degree of initial qualification rigor if warranted. This is in keeping with the goal that MRP-228 continue as a living document with frequent updates as new or improved transducer delivery tooling or transducer packages are added to the experience base as dictated by industry needs and lessons learned. The technical justification process described in ASME Section V Article 14 as incorporated into MRP-228 allows for these gains or adjustments to be made in an open, standardized manner.

Thus MRP suggests the following re-wording of this section:

An extensive experience base for the aging management methodologies described in this section of the I&E guidelines permits selection of known aging management

methodologies. Many inspections specified herein are remote visual examinations whether visual VT-1, EVT-1 or VT-3. For remote visual no procedure qualifications are required other than that in addition to ASME Code Section XI requirements remote visual examinations must meet the additional generic requirements of MRP-228 for equipment and training of personnel, and in the case of visual EVT-1 a surface condition assessment, and limitations on camera angle and scan speed. All other methodologies specified herein already have well established procedural qualifications, such as volumetric examination of bolting. Thus the level of procedural qualification for examinations other than remote visual is limited to technical justification. This level of qualification is appropriate. Failures of internals do not result in pressure boundary failures. Internals are either of robust design resulting in flaw tolerance well above the detection level that can be established via technical justification or consist of assemblies for which single (or often multiple) component item failures do not prevent the assembly from performing its function. The Inspection Standard [3] provides detailed guidance for conducting and justifying the selected examination techniques and the technical justifications required for different examination methodologies and component configurations.

RAI 3-11 Section 4.2.5 of TR MRP-227, Rev. 0, discusses the criteria when particular aging effect indications need to be coupled to physical measurement methods. However, the discussion in Section 4.2.5 does not prescribe the physical measurement methods or techniques that would be used to quantify these aging effect indications.

Please amend Section 4.2.5 to specify the physical measurement techniques that will be used to quantify the aging effect indications for which they are credited.

Response: The particular techniques to be utilized for physical measurements are not within the scope of MRP-227, but are covered generically by MRP-228 through the statement that a technical justification is required for any examinations other than visual examinations. Sections 4.3.1 and 4.3.3 of MRP-227 describes the physical measurements needed for the B&W internals core clamping items and the Westinghouse internals hold-down spring, respectively. In addition, Tables 4-1 and 4-3 provide the required examination methods and examination coverage and Tables 5-1 and 5-3 provide the acceptance criteria for the physical measurements. Physical measurement is specified in Table 4-2 for the CE units with core barrel shroud assembled in two vertical sections and a gap between the top and bottom core shroud segments is identified first by VT-1. Also, refer to the response to RAI 3-12 for the Westinghouse internals hold-down spring.

RAI 3-12 Section 5.2 indicates that acceptance criteria for physical measurement techniques for W-designed RVI components are not included in TR MRP-227, Rev. 0, because the tolerances are available on a plant-specific or design-specific basis. Clarify that the intent of this statement is that licensees of W-designed facilities must obtain acceptance criteria for specified physical measurements based on information in their plant's CLB. Hence, the physical measurements taken during as part of the plant's license renewal AMP must demonstrate that the condition of the affected component remains consistent with the plant's current licensing basis. Identify that this would be an applicant action item for W designed facilities who plan to implement TR MRP-227, Rev. 0.

Response: Licensees of Westinghouse-designed facilities would need to obtain acceptance criteria for specified physical measurements based on information in their plant's Current Licensing Basis. Section 5.2 applies specifically to type 304 hold-down springs in Westinghouse designed plants. The inspection is intended to demonstrate that the hold-down spring has sufficient remaining compressibility to maintain the hold-down forces on the internals. Required hold-down forces are part of the current licensing basis. Calculation of the spring height required to maintain the hold down forces and determination of appropriate margin against further deformation prior to the next scheduled inspection would be the plant responsibility. This determination would be part of the pre-inspection engineering program implemented by the utility.

RAI 3-13 Clarify whether the acceptance criterion for eddy current (ET) inspections is based on a "pass – no pass" acceptance criterion (i.e., any ET signals indicating a relevant ET indication would fail the acceptance criterion).

Response: This RAI is very similar to RAI 2-3. Therefore, the response will be similar. Section 4.2.3 of MRP-227 specifically identifies eddy current surface examination as an electromagnetic testing (ET) method that can be used to supplement visual examination methods, in order to further characterize any detected relevant indications. As indicated in RAI 3-10 response, MRP-228 requires technical justifications for qualification of NDE systems other than visual examinations.

Eddy current examination is called out in Table 4-9 as an existing program carried out for flux thimble tubes in Westinghouse plants, in accordance with plant commitments made in response to NRC I&E Bulletin 88-09. In the latter case, the acceptance criteria for any detected indications are part of the plant commitments. All other applications of eddy current examinations will be supplemental examinations and, as such, this examination technique is not one of the prime examination methods for which specific examination acceptance criteria are required. When eddy current surface examination is used to supplement visual examination, the purpose will not be to again identify the relevant condition, but instead to further characterize the indication by – for example – confirming the crack-like nature of the indication and more accurately sizing its surface-breaking length. In such a case, the acceptance criteria to be applied will not be *examination* acceptance criteria, but *evaluation* acceptance criteria. These *evaluation* acceptance criteria are referred to in the context of supplementary examinations, engineering evaluations, and repair/replacement in Section 6 of MRP-227 (evaluation acceptance criteria have been developed for the MRP by the PWR Owners Group[3], and will be provided to NRC by the MRP).

RAI 3-14 Bolts in some RVI components may be subject to stress relaxation resulting in reduction in preload due to thermal and irradiation effects and, as such, are inspected at every 10 year interval under the ASME Code, Section XI, ISI program. During this interval, reduction in preload in these bolts should not be large enough to cause loss of component functionality prior to the next examination. The evaluation of the need to maintain bolt preload should also consider the impact of loss of preload on vibrational fatigue damage to the bolt and/or the component itself.

[3] Westinghouse Non-Proprietary Class 3 Report, "*Reactor Internals Acceptance Criteria Methodology and Data Requirements, WCAP-17096-NP, Revision 2*" December 2009

Explain (based on the minimum number (percentage) of bolts/springs in each component that are required to maintain preload) how the proposed 10 year frequency is adequate to maintain functionality of each component under all design basis conditions.

Response: Past operating experience and the results from previous PWR internals bolted assembly examinations have shown that loss of preload is unlikely to change significantly during 10-year ISI intervals. For example, inspections and bolt replacements at European reactors have shown a mixture of preload relaxation and additional loading. In addition, inspection of the baffle-to-former bolts in the internals at least two U.S. units has not identified any failed bolting, either from IASCC or from vibrational fatigue, at a point in operational time when both relaxation and/or reloading of the bolts is expected to have occurred.

This experience base has been confirmed by functionality analyses performed on bolted assemblies in support of the development of the MRP-227, Revision 0 inspection and evaluation guidelines. These analyses clearly show that some bolt locations can be expected to have significant loss of preload, in some cases with loss of preload of 100%; others in less highly irradiated locations can be expected to have very little loss of preload; and others can be expected to initially have loss of preload, followed by eventual re-loading as the result of void swelling and possibly prying action from connected structure. It should be pointed out that these functionality analyses were limited to long-term, steady-state operation (including neutron irradiation and gamma heating), but did not encompass the entire spectrum of loading conditions. In spite of the limitations, the functionality analyses illustrate the variation in bolting response over time very well. The analyses show loss of preload due to irradiation and overall structural response begins early in plant life. A substantial fraction of the total expected loss of preload occurs in the first 20 years. Subsequent loss of preload is limited and gradual.