Request for Additional Information
EPRI Technical Report No. 3002005594, BWRVIP-25, Revision 1,
"BWR Core Plate Inspection and Flaw Evaluation Guidelines"

By letter dated September 26, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16273A474[1]), the Electrical Power Research Institute (EPRI) Boiling Water Reactor (BWR) Vessel and Internals Project (BWRVIP) submitted EPRI Technical Report No. 3002005594, "BWRVIP-25, Revision 1: BWR Vessel and Internals Project, BWR Core Plate Inspection and Flaw Evaluation Guidelines," for U.S. Nuclear Regulatory Commission (NRC) staff review. BWRVIP-25, Revision (Rev.) 1 provides a set of augmented inspection and evaluation (I&E) criteria that may be used to either inspect or evaluate the reactor vessel internal core plate assemblies that are present in BWR plant designs. BWRVIP-25 represents an update of the previous I&E guidelines for the core plate assemblies in EPRI Topical Report (TR) No. 107284, "BWRVIP Vessel and Internals Project, BWR Core Plate Inspection and Flaw Evaluation Guidelines (BWRVIP-25)," which was accepted for use in an NRC staff-issued safety evaluation (SE) dated December 19, 1999 (ML993620274).

Based on its review of BWRVIP-25, Rev. 1, the NRC staff has determined that additional information is needed to complete its review. The NRC staff's request for additional information (RAI) is provided below. This NRC RAI contains information that is considered proprietary by EPRI. All EPRI proprietary text is marked in [[ ]].

MVIB Operating Plants RAI-1: Performance of Core Plate Bolt Inspections for BWR Plants during Periods of Extended Operation

Background: For all operating U.S. BWR plants that do not have lateral restraint wedges, Section 3.2.2 of the original BWRVIP-25 report (BWRVIP-25, Rev. 0, December 1996) identifies that the core plate bolts [[ ]] In the course its review of BWR plants license renewal (LR) applications (LRAs) for 20-year extended license terms (periods of extended operation (PEOs)), the NRC staff has approved BWR renewed license holders use of the original BWRVIP-25 report as a basis for aging management of the core plate, which includes [[ ]]

Renewed licenses were generally conditioned to require implementation of aging management activities that are described in the Final Safety Analysis Report (FSAR) sections for LR (including BWRVIP I&E guidelines for the BWR internals) during PEOs. However, for all BWR plants [[ ]], it was established that implementation of the [[ ]] is not feasible for inspecting the core plate bolts and therefore, BWR plants [[ ]] required a deviation from the BWRVIP-25 inspection guidelines. These BWRVIP-25 deviation letters included some limited justification for not performing the [[ ]] and were submitted to the NRC for information only, without any regulatory requirement for NRC staff review.

[1] Henceforth, documents that are identified in this report and tracked in the staff's ADAMS will be designated solely by their accession number designations.

BWRVIP-25, Rev. 1, Section 3.1.1, page 3-3 states that 71 core plate bolt inspection evolutions have been performed at 21 plants that are General Electric (GE) Type 2 through 5 BWR containments (BWR/2-5s). Some inspections were performed on a 100-percent basis, while others were performed on a percentage or an as-accessible basis. Specifically, "fifteen of the inspections were performed using Visual Testing (VT)-1, Enhanced Visual Testing (EVT)-1, or Modified Visual Testing (MVT)-1 methods, 51 were performed using the VT-3 method, and five were performed by confirming the presence of bolts using UT methods on adjoining structures…In all cases no indications were observed." The TR identifies that [[

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Issue: Based on the statements provided in BWRVIP-25, Rev. 1, Section 3.1.1, page 3-3, the NRC staff has inferred that some BWR plants have been able to perform either, VT-1, EVT-1, or MVT-1 examinations of the core-plate (CP) bolts, and many BWR plants have been able to perform VT-3 exams of the bolts. In addition, during the LRA reviews, some BWR renewed license holders had made commitments for LR to perform analytical evaluations for demonstrating that the integrity and functionality of CP assembly would be maintained during PEOs.

These LR commitments, which often got incorporated into the FSAR, specified that a plant-specific stress analysis of the CP assembly would be performed, taking into consideration the loss-of-bolt preload due to stress relaxation from irradiation and thermal effects, as well the potential for bolt cracking during the PEO. The analysis would be submitted to the NRC staff for review. Through its review of these plant-specific analyses, the NRC staff has requested that licensees commit to performing VT-3 visual examination of a 50 percent sample of the CP bolts during PEOs in order to provide reasonable assurance that the bolts and their locking devices are remaining in place during the PEO.

Some BWR licensees have committed to performing the VT-3 exams as an aging management activity. These regular commitments were made during the course of the NRC staff review of the plant-specific CP analytical evaluations for closure of the original LR/FSAR commitments. For an example, please see the October 2013 supplemental RAI response provided by Nebraska Public Power District for Cooper Nuclear Station at ML13283A010.

Request: Based on the above-cited past experience and precedent for performing visual examination of accessible CP bolts for the detection of significant degradation, please provide the following information regarding future CP bolt inspection criteria, to include inspection method, frequency, and inspection sample size, that will be conducted during PEOs for the following categories of BWR plants:

1. Those plants that satisfy the evaluation criteria specified in BWRVIP-25, Rev. 1, Appendix I, Section 9.7 for elimination of CP bolt inspections. Specifically, for these plants, please discuss whether any in-vessel visual inspections would be conducted to provide reasonable assurance that the bolts and their locking devices are remaining in place during PEOs. Please revise and/or supplement Appendix I to address performance of these core bolt inspections for BWRs seeking to implement the Appendix I methodology.

2. Those plants that do not satisfy the evaluation criteria specified in Appendix I shall require a plant-specific justification and/or alternative, as specified in Section 9.7.

BWRVIP-25, Rev. 1, Appendix A provides an example for a plant-specific CP bolt stress analysis that would need to be performed if the plant fails to meet the evaluation criteria in Appendix I, Section 9.7. Appendix A identifies that [[

.]] Please address how the plant-specific CP bolt inspection criteria will be determined based on the results of this plant-specific analysis. Please revise and/or supplement BWRVIP-25, Rev. 1, Appendix A to address the determination of core bolt inspection criteria for BWRs that need to perform this plant-specific analysis.

3. For those plants that do not satisfy the Appendix I evaluation criteria, and for which a plant-specific stress analysis does not demonstrate acceptable margins, per the example provided in Appendix A, please identify whether these plants would be required to perform [[
]]. Please revise and/or supplement the BWRVIP-25, Rev. 1 to address performance of these CP bolt inspections.

MVIB Operating Plants RAI-2: Inservice Inspection (ISI) of the Core Plate, per the American Society of Mechanical Engineers (ASME) Code, Section XI, Examination Category B-N-2

Background: The first paragraph of BWRVIP-25, Rev. 1, Section 3.1.1 identifies that most plants include inservice examination of the CP under the ASME Code, Section XI, Examination Category B-N-2, Item No. B13.40 for welded core support structures (CSSs). Examination Category B-N-2 of the ASME Code, Section XI requires VT-3 exams of “accessible surfaces” of welded CSSs. The third paragraph of Section 3.1.1 states that the ASME Code, Section XI, Examination Category B-N-2, Item No. B13.40 “accessible surfaces” phrase is clarified to be those areas “made accessible for examination by removal of components *during normal refueling outages*” [emphasis added]. This third paragraph further states that during a typical refueling outage, “the shuffling of fuel bundles does not allow access to the core plate,” and for this reason, “most plants consider core plate subcomponents inaccessible for examination,” based on the ASME Code, Section XI, Examination Category B-N-2 ISI requirements.

Issue: The NRC staff identified that the above statements are not consistent with the later BWRVIP-25, Rev. 1, Section 3.1.1 statements addressing performance of visual exams. Specifically, Section 3.1.1, page 3-2 states that [[

.]] Page 3-2 of TR, Section 3.1.1 then indicates that SIL 588, Rev. 1 recommended that [[

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Request:

1. Please reconcile the above two contradictory statements regarding the accessibility of the CP bolts for VT-3 visual examination.

2. The NRC staff notes that VT-3 examination of “accessible surfaces” of the CP once every 10-year ISI interval is required by the ASME Code, Section XI, Examination Category B-N-2. Based on the above BWRVIP-25, Rev. 1, Section 3 statemen, it is

unclear how plants proposing to implement the new guidelines will perform the ASME Code, Section XI ISI of the core plate in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 50.55a. Please address whether plants proposing to implement this BWRVIP-25, Rev. 1, methodology will ensure compliance with either (i) the ASME Code, Section XI, Examination Category B-N-2 requirement for VT-3 examination of accessible surfaces of the core plate during refueling and/or maintenance activities; or (ii) plant-specific alternatives authorized by the NRC staff pursuant to 10 CFR 50.55a(z)(1) for implementation of BWRVIP-25 guidelines in lieu of the ASME Code, Section XI, Examination Category B-N-2 VT-3 examinations.

MVIB and ESEB Operating Plants RAI-3: IGSCC Mitigation and Evaluation of IGSCC & Fatigue Cracking for the Core Plate Bolt Structural Analysis in TR Appendix I

<u>Background</u>: 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 1, "Quality standards and records," requires structures, systems, and components be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. GDC 2, "Design bases for protection against natural phenomena," requires structures, systems, and components important to safety to be designed to withstand the effects of natural phenomena, such as earthquakes, without loss of capability to perform their safety functions. In accordance with 10 CFR 54.21(a)(3), aging management programs are specifically required to ensure that the effects of aging on structures and components will be adequately managed during PEOs so that intended functions are maintained consistent with the current licensing basis (CLB), which includes GDCs. Renewed licenses were generally conditioned to require implementation of aging management activities that are described in the plants' FSAR sections for LR (including BWRVIP I&E guidelines for the BWR internals) during PEOs in order to ensure that safety functions are maintained consistent with CLB requirements.

<u>Issue</u>: BWRVIP-25, Rev. 1, Appendix I, Section 4.3 states that [[ ]] Hydrogen Water Chemistry (HWC) [[ ]] Noble Metal Chemical Addition (NMCA) [[ ]] in order to use the Appendix I methodology for structural analysis of the CP bolts. Furthermore, since CP bolts in U.S. BWRs have not been volumetrically examined in accordance with the original BWRVIP-25 inspection guidelines, the NRC staff considers the extent of cracking in the CP bolts to be unknown. However, the Appendix I methodology for structural analysis of the bolts seems to be predicated on the assumption that core plate bolt cracking (due to intergranular stress corrosion cracking (IGSCC)) would not occur, based exclusively on an evaluation of the bolt fabrication method, which is discussed in Section 4 of Appendix ; bolt fabrication method alone would not totally preclude IGSCC in a sufficiently oxidizing environment. Additionally, BWRVIP-25, Rev. 1 is silent on the potential for CP bolt cracking due to other aging mechanisms, like fatigue.

Therefore, the NRC staff determined that it currently does not have adequate assurance that the CP bolts would be resistant to IGSCC and fatigue cracking. Accordingly, the NRC staff cannot evaluate the validity of the BWRVIP-25, Rev. 1, Appendix I methodology as a basis for aging management for ensuring the structural integrity and functionality of the CP bolts, consistent with CLB requirements in GDC 1 and GDC 2 during PEOs, without an evaluation of either: (1) how the loss of CP bolt functionality as a result of IGSCC and fatigue cracking is considered as a specific input into the Appendix I structural analysis; or (2) how IGSCC and fatigue cracking would be specifically considered later by plants seeking to use the Appendix I structural analysis methodology to demonstrate acceptable structural margins.

Request:

1. For normal water chemistry (i.e., no credit for HWC and NMCA/OLNC), please address how the loss of CP bolt functionality as a result of IGSCC and fatigue cracking is evaluated for determining a bounding number and distribution (i.e., clustering) of failed bolts for the BWRVIP-25, Rev. 1, Appendix I structural analysis. This technical justification should specifically address how the following attributes are incorporated into the Appendix I structural analysis to determine the minimum number and bounding distribution of intact (crack free) bolts that are necessary to satisfy the structural acceptance criteria: (1) the extent of IGSCC in the bolts; (2) the distribution of IGSCC in the bolts (i.e., randomness or clustering of cracking in various locations); (3) the effects of fatigue cracking on the bolts; and (4) the effects of potential clustering of cracked and non-functional bolts on the stress analysis and worst-bolt determination used in the parametric study, including a consideration of moments and stress conditions generated by asymmetrical or eccentric clustering of non-functional bolts.

2. Plant-specific application of the BWRVIP-25, Rev. 1, Appendix I structural analysis must provide reasonable assurance that the CP bolts will maintain their functionality to ensure safe-shutdown capability under seismic and loss-of-coolant accident (LOCA) loadings during PEOs, specifically considering the potential for a bounding number and distribution (i.e., clustering) of cracked and non-functional bolts, based on the occurrence of IGSCC and fatigue. Therefore, for plants with normal water chemistry (no credit for HWC and NMCA/OLNC), if the effects of IGSCC and fatigue were not already considered for the Appendix I structural analysis, please revise and/supplement BWRVIP-25, Rev. 1, Appendix I to address how applicable plants using the Appendix I, Section 9 evaluation process will specifically determine whether they have an acceptable number and distribution of intact bolts to satisfy the structural-acceptance criteria, based on a conservative plant-specific calculation of a certain number of non-functional bolts due to IGSCC and fatigue cracking.

MVIB Operating Plants RAI-4: BWRVIP-25, Rev. 1, Appendix A – Consideration of Core Plate Bolt Aging Effects

Background/Issue: BWRVIP-25, Rev. 1, Appendix A provides an example for the CP bolt stress analysis if a plant fails to meet the Appendix I evaluation applicability criteria. Appendix A addresses determination of loadings and the calculation of stresses on the CP bolts. However it does not specify how reduction in bolt preload due to stress relaxation (per the mechanisms identified in Appendix I) and the potential for bolt cracking would be accounted for in a plant-specific analysis.

Request: Please revise and/or supplement BWRVIP-25, Rev. 1, Appendix A to address the effects of stress relaxation and cracking for the core plate bolts.

MVIB Operating Plants RAI-5: IGSCC Susceptibility Based on Core Plate Bolt Fabrication and Procurement Specification

Background: As discussed above, the NRC staff has previously reviewed licensee submittals in fulfillment of LR/FSAR commitments for demonstrating CP bolt functionality during PEOs. As discussed above, certain licensees made regular commitments for perform VT-3 exams of the bolts as an aging management activity. These commitments were made as part of the staff's review of the CP bolt analytical evaluations for demonstrating CP bolt functionality. The NRC staff identified that its approval of these plant-specific CP bolt analyses and associated commitments to perform VT-3 exams was based, in part, on its determination that the licensees adequately demonstrated that their CP bolts would have a low susceptibility to IGSCC. These NRC staff findings are documented in the following correspondence:

- Section 3.2.2 of the March 28, 2012, SE for the Vermont Yankee (VY) CP hold-down bolt inspection plan and stress analysis for closure of a LR Commitment (ML120760152);

- Section 3.2.3 of the July 25, 2014, SE for the Cooper CP hold-down bolt inspection plan and stress analysis for closure of a LR Commitment (ML14190A004).

The NRC staff's determination regarding the low IGSCC susceptibility for the CP hold-down bolts at VY and Cooper were based on the fact that the VY and Cooper CP bolts are not sensitized. The bolts were procured to a specification prohibiting cold forming operations after solution heat treatment, and there were no known instances of stress corrosion cracking (SCC) in these bolts in the BWR fleet at that time.

Issue: BWRVIP-25, Rev. 1, Appendix I, Section 4.2 states that all [[ ]]

BWRVIP-25, Rev. 1, Appendix I, Section 4.2 also states that [[.]]

Request:

1. For all plants listed in Table 3-1 of BWRVIP-25, Rev. 1, Appendix I, please identify whether the original bolt procurement specification specifically required the [[ ]] bolt material to be solution heat treated following the cold roll threading process.

2. In addition, for all plants listed in Table 3-1 of Appendix I, please identify whether the original bolt procurement specification also limited the as-fabricated material surface hardness to be below a certain value in order to limit the amount of [[ ]] cold work introduced as part of the [[ ]].

MVIB Operating Plants RAI-6: Thermal Stress Relaxation for Core Plate Bolting

BWRVIP-25, Rev. 1, Appendix I, Section 6.2 identifies that small amounts of plastic deformation due to mechanisms associated with thermal creep would result in a [[ ]] reduction in bolt preload. The basis for this value is References 11 and 12 of the BWRVIP-25, Rev. 1. Please discuss how this value was calculated and address how it is bounding for all Appendix I, Table 3-1 BWR plants.

MVIB Operating Plants RAI-7: Irradiation-Enhanced Stress Relaxation and Neutron Fluence Evaluation for Core Plate Bolting

<u>Background/Issue</u>: Based on the stress relaxation evaluation described in BWRVIP-25, Rev. 1, Appendix I, Section 6.3, the NRC staff identified that the amount of projected stress relaxation due to neutron irradiation for 60 years of operation is [[ ]].

However, the BWRVIP-25, Rev. 1 does not provide detailed calculations of these values for demonstrating that they are bounding for all BWR plants listed in Table 3-1 of BWRVIP-25, Rev. 1, Appendix I. Nor does it address how the neutron fluence values that were used to calculate the projected stress relaxation due to irradiation were determined to be bounding for all BWR plants listed in Table 3-1.

Request:

1. Please discuss how the projected stress relaxation values due to neutron irradiation were calculated and address how they are bounding for all Appendix I, Table 3-1 BWR plants, taking into consideration the differences in plant-specific CP bolt configuration and geometry.

2. Please address how the Appendix I, Section 6.3 neutron fluence values that were used as the basis for determining projected decrease in CP bolt preload due to irradiation-enhanced stress relaxation were determined to be bounding for the BWR plants in Appendix I, Table 3-1, taking into consideration variation in neutron flux as a function of bolt azimuthal location around the periphery of the core plate and differences in plant-specific neutron fluence for the bolts.

MVIB Operating Plants RAI-8: Neutron Fluence Methodologies for Core Plate Bolting

Background: BWRVIP-25, Rev. 1, Appendix I, Section 6.3 references [[

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<u>Issue</u>: The NRC staff identified that these neutron fluence methodologies were approved by the NRC staff only for the specific applications identified therein – specifically, reactor-pressure vessel (RPV) integrity evaluations.

<u>Request</u>: Please address how these methodologies were validated for calculating the specific neutron fluence values identified in Section 6.3, taking into consideration any benchmarking of the calculations (based on measured neutron activation of material samples) for application to core plate bolting.

ESEB Operating Plants RAI-1: Appendix I Structural Analysis

<u>Background</u>: GDC 1 requires structures, systems, and components be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. GDC 2 requires structures, systems, and components important to safety to be designed to withstand the effects of natural phenomena, such as earthquakes, without loss of capability to perform their safety functions. ASME Code Section III,

Subsection NG, Table NG-3352-1 provides the a tabulation of appropriate weld quality factors based on the type of welded joint, and the type of examination(s) performed.

Issue: BWRVIP-25, Rev. 1, Appendix I, Section 8.3 states that [[

]] Sufficient technical justification is not provided to justify the use of a weld quality factor of [[ ]], as the strength of unmodeled welds cannot be credited or exchanged for an increase in weld quality factor.

Request: Please either reduce the specified weld quality factor to [[ ]], or provide a description of the modeled welds that demonstrates the pedigree required for the use of the specified weld quality factor, or revise the model to include unmodeled welds as needed to provide sufficient margin.

MVIB License Renewal RAI-1

Background: BWRVIP-25, Revision 1, Appendix B, Section B.1 identifies that BWR core plates will need to be within the scope of an LRA or a subsequent LRA (SLRA) because they serve intended functions needed to either: (a) shut down the reactor and maintain it in a safe-shutdown condition, as defined in 10 CFR 54.21(a)(1)(ii), or (b) prevent or mitigate the consequences of design basis accidents, as defined in 10 CFR 54.21(a)(1)(iii).

Issue: BWRVIP-25, Revision 1, Appendix B, Section B.1 does not indicate whether the core plate rim hold-down bolts or CP wedge restrainers (as applicable and relied on for protecting the core plates against lateral movements) will need to be included in the scope of an LRA or SLRA, as required by either 10 CFR 54.4(a)(1)(ii) or (iii), or in accordance with 10 CFR 54.4(a)(2), which applies to the scoping of non-safety related components whose failures could impact the intended function(s) of a safety-related structure or component serving a reactor coolant pressure boundary, safe shutdown, or accident mitigation intended function.

Request: Clarify whether BWR CP rim hold-down bolts or core plate wedge restrainers will need to be included in the scope of an LRA or SLRA under the requirements of 10 CFR 54.21(a)(1)(ii) or (iii) or in accordance with the scoping requirements for non-safety related components in 10 CFR (a)(2). Justify the basis for your response.

MVIB License Renewal RAI-2

Background: BWRVIP-25, Revision 1, includes Appendix B, "Demonstration of Compliance with the Technical Information Requirements of the License Renewal Rule (10 CFR 54.21)." On page B-4 of the report, EPRI states that crack initiation and growth is the only aging effect for the core plate that requires an aging management review (AMR) for license renewal.

Issues: 1. EPRI's statement implies that the need for subjecting a structure or component (SC) to an AMR is limited only to those components that have one or more aging effects requiring management (AERMs). This is not consistent with the requirements in 10 CFR 54.21(a)(1) – (a)(3). The rule requires a given SC to be the subject of an AMR if they are not active or involve moving parts or configuration and if they are not subject to replacement based on a qualified life or specified time frame (i.e., passive, long-lived SCs). For those SCs that are determined to be passive, long-lived SCs, the requirements in 10 CFR 54.21(a)(1) would require a given SC to be subject to an AMR even if there were no AERMs attributed to the material-environmental combination for the SC.

2. Many past LRAs for BWR facilities have identified loss of preload due to thermal or irradiation-enhanced stress relaxation as an AERM for their BWR core plate rim hold-down bolts. To be consistent with past practices, the NRC staff identified this AERM in AMR line item IV.B1.R-420 of Table IV.B1 in the GALL-SLR report (i.e., Table IV.B1 in NUREG-2191, Volume 1).

3. In some past LRAs for BWR plants, the LRAs identified that cumulative fatigue damage or cracking due to fatigue or cyclic loading is an AERM for CP assemblies or specific CP assembly components that were within the scope of the LRAs.

<u>Requests</u>:

1. Justify why the report's methodology limits SCs subject to AMR only to those that have applicable aging effects.

2. Justify why loss of preload due to stress relaxation or irradiation assisted creep is not identified as an AERM for the core plate rim hold-down bolts.

3. Provide the basis why BWRVIP-25, Rev. 1 does not identify cumulative fatigue damage or cracking due to fatigue or cyclic loading as an applicable AERM for BWR core plate assemblies and assembly components.

MVIB License Renewal RAI-3

<u>Background</u>: BWRVIP-25, Rev 1, Appendix B, Section B.3 identifies that some CP assemblies are designed with wedge restrainers in the assembly design. EPRI made the following proprietary statement with respect to these types of CP assemblies:

"[[

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Issue: EPRI's inspection basis for CP assembly designs that rely on wedge assemblies to secure the core plates was consistent with the CLBs for past BWR LRAs whose core plate assemblies were restrained with wedges. EPRI's statement (above) leaves [[

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This type of AMR basis creates a regulatory issue for core plate assembly designs that are secured with wedge restrainers because it may imply that the wedges may not be reliable for restraining the core plates if the loadings on the wedge assemblies were to exceed upper bound acceptance limits on design basis stress levels or stress intensity values.

However, the scope of BWRVIP-25, Rev. 1 does not include any generic technical stress evaluation appendix for CP assembly designs that utilize wedge restrainers, such that the upper bound limits on the allowable stress loadings or stress intensity factors for the wedge restrainers would be firmly established in the BWRVIP-25, Rev. 1 report. Thus, the NRC staff questions how an applicant for a LRA or SLRA would be capable of performing this type of confirmatory action when BWRVIP-25, Rev. 1 fails to include any bounding generic stress analysis for assembly designs that utilize and rely on wedge restrainers as the basis for securing the core plates during design basis loading conditions.

Request: Justify the basis for omitting a structural analysis report appendix in BWRVIP-25, Rev. 1, for those core plate designs that are restrained with wedges and why the report does not firmly establish the upper bound limits for stresses, loads, or stress intensities associated with the design basis loading conditions of the wedge restrainers in the core plate assembly designs. Clarify whether there could be any AERMs in the wedge restrainers if the stress loads associated with the components were to exceed the upper bound stress or stress intensity limits set in the stress analysis for the wedge restrainers. If so, identify the AERMs associated with those components that may need to be managed during the period of extended operation (including subsequent periods of operation for proposed SLRAs).

If there are AERMs, define and justify the corrective actions a BWR would take under its BWR Vessel Internals Program to manage the AERMs that may be manifested if the maximum allowable stress levels or stress intensity factors for the wedge restrainers were to be exceeded. Lastly, justify why the action requesting verification of the structural analysis has not been identified as an applicable license renewal applicant action item for the BWRVIP-25, Rev. 1 methodology.

MVIB License Renewal RAI-4

Background: BWRVIP-25, Rev. 1, Section B.3(c) states that crack initiation and growth will be managed by an inspection program that incorporates the inspection guidance provided in Section 3.0. However, BWRVIP-25, Rev. 1, Appendix I provides a general, time-dependent bolt stress relaxation methodology that may be used as an alternative to the criteria for inspecting BWR core plate rim hold-down bolts (CPRH-DBs). BWRVIP-25, Rev. 1, Section 3.2.2.2 and Appendix I, Section I.1.2 state that "good inspection results combined with the good operating experience of BWR bolts and the degree of redundancy of the core plate bolts may justify elimination of any reinspection."[2] Section I.1.2 further states that the evaluation in Appendix I "provides justification for the elimination of inspection of CPRH-DBs if the plant meets the minimum acceptability requirements of Section 9.7" of Appendix I.

Issue: EPRI's basis for allowing use of the Appendix I methodology appears to rely on the general assumption that there has not been any operating experience (OpE) with cracking of US. BWR CPRH-DBs to date, or if it has occurred, that the amount and extent of cracking in the bolts is minimal. EPRI does not define which type of bolting is being referenced in the terminology "good operating experience with BWR bolting," and what EPRI means by the statement "good operating experience." Even if there has been good OpE with other types of BWR bolting, the OpE may not be indicative and representative of the material condition in BWR CPRH-DBs, at least not without citing and summarizing appropriate baseline inspection results of BWR CPRH-DBs to support such a conclusion.

As a minimum, baseline inspection results from a reasonable sample of past inspections performed on U.S. BWR CPRH-DBs would be needed to support a conclusion that, in all probability, cracking has not occurring in a plant's CPRH-DBs or is minimal. Yet many past BWR LRA applicants have identified in their previous LRAs that they cannot perform the BWRVIP-defined inspections of their CPRH-DBs due to accessibility issues with the configurations of the core plate assemblies at their facilities. Also EPRI has yet to provide any past CPRH-DB inspection data to support its assumptions on this matter. In addition, BWRVIP-25, Rev. 1, Appendix B, Section B.3.(c) fails to include any statement that the alternative stress relaxation analysis methodology in BWRVIP-25, Rev. 1, Appendix I may be used to eliminate future inspections of BWR CPRH-DBs. Thus, additional information is need to resolve these issues.

Request: Clarify whether use of the methodology in Appendix I is predicated on an assumption that there has been no past OpE with cracking in BWR CPRH-DBs, or that the amount of cracking is minimal. Provide the CPRH-DB inspection data that supports this conclusion. If there is no supporting inspection data, justify why it would be permissible for a BWR license renewal applicant to use the methodology in Appendix I as a basis for eliminating future inspections of its BWR CPRH-DBs. Justify why BWRVIP-25, Rev. 1, Section B.3(c) does not address this possibility as a specified alternative to the performance of UT or enhance visual inspections of the CPRH-DBs.

MVIB License Renewal RAI-5

Background: BWRVIP-25, Rev. 1, Appendix B establishes how the BWRVIP-25, Rev. 1 may be used to comply with requirements in 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants." BWRVIP-25, Rev. 1, Appendix I provides a generic evaluation methodology that may be used as an alternative to the criteria for inspecting BWR CPRH-DBs in BWRVIP-25, Rev. 1, Section 3. Specifically, Appendix I, Section 9.7 provides the criteria that need to be met to justify use of the appendix for elimination of the inspection protocols for the assemblies.

The NRC staff's aging management program (AMP) for inspecting BWR CPRH-DBs is provided in AMP XI.M9, "BWR Vessel Internals," as included in NUREG-1801, Revision 2 (i.e., the Generic Aging Lessons Learned [GALL] Report) for LRAs, or NUREG-2191, Volume 2 (GALL-SLR) for SLRAs. For aging management of BWR CP assemblies, the AMP invokes the inspection methods previously approved for these types of assemblies in BWRVIP-25.

The NRC staff-endorsed guidance in the Nuclear Energy Institute (NEI) guidance NEI-95-10, Revision 6, "Industry Guidelines for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule," (ML051860406), provides the industry's main guidance methodology for the format and contents of LRAs that are required to be submitted in accordance with the 10 CFR Part 54 rule. NEI 17-01, "Industry Guidance for Implementing the Requirements of 10 CFR Part 54 for Subsequent License Renewal," provides the analogous criteria for SLRAs. The NEI guidance documents define when alternative aging management criteria proposed by license renewal applicants would need to be identified as exceptions to the stated program element criteria in GALL-based or GALL-SLR-based AMPs.

Issue: Since AMP XI.M9 has yet to reference use of BWRVIP-25, Rev. 1, the AMP does not identify that use of the evaluation methodology in BWRVIP-25, Rev. 1, Appendix I is an acceptable alternative to the performance of augmented inspections of BWR CPRH-DBs.

Request: Clarify the additional criteria and justifications a BWR applicant will need to identify and incorporate into the BWR Vessel Internals Program of its LRA or SLRA in order to justify use of the BWRVIP-25, Rev. 1 report as the basis for managing aging in the CP assembly and CP assembly components of its reactor design. Include all inspection-based or analytical-based options that LRA or SLRA applicant may use to manage the effects of aging that are applicable to passive, long-lived components in the core plate assemblies.

MVIB License Renewal RAI-6

Background: The BWRVIP-25, Rev. 1 report includes Appendix A, "Example Core Plate Bolt Analysis." For BWR plants relying on bolts for the integrity of their core plates, Appendix A indicates that it is provided as "an example for the plant-specific core plate bolt stress analysis if a plant fails to meet the application criteria to eliminate the requirements of the inspection of the of the core plate bolts specified in Appendix I" of the report.

In contrast, BWRVIP-25, Rev. 1, Appendix B makes the following proprietary statement regarding inspection strategies for CPRH-DBs and the implementation of plant-specific stress analyses for the bolts:

[[

]] [1 2]

Issue: The various statements referenced in the background section above create confusion on the specific types of circumstances that would prompt a BWR license renewal applicant to perform a plant-specific bolt stress analysis in accordance with the methodology in BWRVIP-25, Rev. 1, Appendix A. The statement in Appendix A implies that a plant-specific bolt stress analysis would only need to be performed if a BWR license renewal applicant had performed a plant-specific stress relaxation analysis assessment of the bolts in accordance with methodology in Appendix I and had failed to meet the acceptance criteria of the evaluation basis in Appendix I. Yet, for those license renewal applicants that may find the inspection bases in BWRVIP-25, Rev. 1, Section 3 feasible for implementation, the statement in Appendix B implies that the licensee or applicant would also need to perform the Appendix A bolt stress analysis in order to establish [[

]][1]

Request: Identify and clarify (with appropriate justifications) all circumstances that would call for a BWR license renewal applicant to perform a plant-specific bolt stress analysis consistent with the methodology in BWRVIP-25, Rev. 1, Appendix A. Factor this into a revision of Appendix A of the report as appropriate.

MVIB License Renewal RAI-7

Background: BWRVIP-25, Rev. 1, Appendix I provides a generic stress relaxation analysis methodology that may be adopted and used to justify elimination of BWRVIP-defined augmented inspections for BWR core plate rim hold-down bolts. Section 6 of Appendix I summarizes the core plate rim hold-down mechanical analysis. The appendix identifies that the analysis involves an assessment of [[ ]] that was based on an assessment of preload loss over a cumulative 60-year licensed plant life. Influences of [[ ]] for the bolts were assessed in Appendix I.

The regulation in 10 CFR 54.21(c)(1) requires license renewal applicants to identify all analyses or evaluations that conform to the definition of a time-limited aging analysis (TLAA) in 10 CFR 54.3(a). In Section 4.1 of NUREG-2192, the NRC staff provided additional clarifications on this matter. The NRC staff identified that analyses, calculations, or evaluations based on 60-year time dependent assumptions would need to be identified as TLAAs in a SLRA if they were determined to conform to the other five criteria for defining TLAAs in 10 CFR 54.3(a).

Issue: Per the criteria in 10 CFR 54.3(a), the stress relaxation analysis in BWRVIP-25, Rev. 1, Appendix I appears to be based on several different time-dependent assumptions that may be defined by current operating term: (a) the time period associated with the assessment of thermally-influenced preload loss, (b) the time period associated with the assessment of preload loss that is influenced by neutron radiation exposure (i.e. neutron fluence exposure), and (c) the time frame for the neutron fluence assessment that factors into the assessment of neutron irradiation-influenced preload loss.

Any BWR licensee performing a plant-specific 60-year Appendix I-based bolt stress relaxation analysis as part of their CLB and intending to use this basis as part the 10 CFR 54.21(a)(3) required basis for managing rim hold-down bolt preload losses in a SLRA, would need to identify and evaluate the analysis as a TLAA for its incoming SLRA, as required by 10 CFR 54.21(c)(1) and use the TLAA as the basis for managing the aging effect under the requirements in 10 CFR 54.21(a)(3). The same concept is valid for those licensees or applicants that have yet to submit a LRA for their BWRs, but had performed an Appendix I-based stress relaxation analysis of the rim hold-down bolts based on a cumulative 40-year plant life.

Yet BWRVIP-25, Rev. 1, Appendix I does not identify the [[ ]][2]. In addition, BWRVIP-25, Rev. 1, fails to include any guidance in Appendices B and I of the report that a plant-specific stress relaxation analysis performed in accordance with the methodology in Appendix I of the report may need to be identified as a plant-specific TLAA for an LRA or SLRA.

Request: Justify why Appendix I does not define the bounding time frame that was used for the [[ ]], consistent with the manner that the EPRI BWRVIP defined the time-frame for this parameter in Section B.4 of Appendix B in the BWRVIP-25, Rev. 1 report. Clarify and justify whether an applicant, that has performed a BWRVIP-25, Rev. 1, Appendix I analysis as part of its CLB, will need to identify the stress relaxation analysis as a TLAA for its SLRA.

MVIB License Renewal RAI-8

Background: BWRVIP-25, Rev. 1, states that crack initiation and growth is the only aging effect for the core plate that requires aging management review for LRAs In past LRAs for BWR-designed plants, many applicants have identified that cumulative fatigue damage or cracking due to fatigue or cyclic loading is an aging effect requiring management for the CP assemblies and have dispositioned this aging effect citing their metal fatigue TLAAs (i.e., cumulative usage factor (CUF) analyses) for the core plates, as given and evaluated in Chapter 4.3 of their LRAs.

Issue: The assessment in BWRVIP-25, Rev. 1, Appendix B, “Demonstration of Compliance with the Technical Information Requirements of the License Renewal Rule (10 CFR 54.21),” does not identify that metal fatigue analyses for the CP assemblies or specific CP assembly components may conform to the definition of a TLAA in 10 CFR 54.3(a) and may need to identified and evaluated as TLAAs in accordance with the requirements in 10 CFR 54.21(c)(1).

Request:

1. Identify all BWR core plate assembly components that have been identified as being within the scope and the subject of an ASME Section III CUF analysis.

2. Justify why BWRVIP-25, Rev. 1, Appendix B does not identify that metal fatigue analyses for core plate assemblies or specific core plate assembly components may need to be identified as applicable TLAAs for LRAs or for subsequent license renewal applications.