Nuclear Operating Company

South Texas Project Electric Generating Station P.O. Box 289 Wadsworth, Texas 77483

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U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

South Texas Project
Units 1 and 2
Docket Nos. STN 50-498, STN 50-499
Response to Request for Additional Information Set 34 for the
Review of the South Texas Project, Units 1 and 2,
License Renewal Application (TAC Nos. ME4936 and ME4937)

References:

1. Letter; G. T. Powell to USNRC Document Control Desk; "License Renewal Application;" NOC-AE-10002607; dated October 25, 2010 (ML103010257)
2. Letter; J. W. Daily to G. T. Powell; "Request for Additional Information Set 34 for the Review of the South Texas Project, Units 1 and 2, License Renewal Application (TAC Nos. ME4936 and ME4937);" AE-NOC-15002760; dated November 30, 2015 (ML15308A014)

By Reference 1, STP Nuclear Operating Company (STPNOC) submitted a License Renewal Application (LRA). By Reference 2, the NRC staff requested additional information (RAI) for their review of the STPNOC LRA. The RAI's contains areas where additional information is needed to complete the review of the license renewal application. STPNOC's response to the RAI's are provided in Enclosure 1 to this letter.

Changes to LRA pages described in Enclosure 1 are depicted as line-in/line-out pages provided in Enclosure 2.

In addition, Enclosures 3 and 4 contain supporting documentation referenced in this submittal.

Regulatory commitment for item 30 in LRA Table A4-1 is revised and depicted as line-in/line-out pages provided in Enclosure 5. There are no other commitments in this letter.

A147
NRR

STI: 34253535

If there are any questions, please contact Arden Aldridge, STP License Renewal Project Lead, at (361) 972-8243 or Rafael Gonzales, STP License Renewal Project regulatory point-of-contact, at (361) 972-4779.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 12-17-15
Date

G. T. Powell

G. T. Powell
Site Vice President

rjg

Enclosures:

1. STPNOC Response to RAI
2. STPNOC LRA Changes with Line-in/Line-out Annotations
3. PWROG-14072-NP, Rev. 0, "South Texas Project Units 1 and 2 Summary Report for the Fuel Design / Fuel Management Assessments to Demonstrate MRP-227-A Applicability," June 3, 2015.
4. PWROG-15001-NP, Rev. 0, "South Texas Project Unit 1 and Unit 2 Summary Report for Applicant/LicenseeAction Items 1, 2, and 7," June 2015.
5. STPNOC Regulatory Commitment

cc:
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Enclosure 1
STPNOC Response to RAI

RAI 3.0.3.3.6-1 - Components within the scope of the AMP

Background:

By letter dated June 30, 2015, the applicant submitted its updated version of the plant-specific PWR Reactor Internals Program (**LRA Section B2.1.35**) for staff review. In the "scope of program" program element for the aging management program (AMP), the applicant identifies that the program includes the following types of components defined for Westinghouse-designed PWRs in the MRP-227-A report: (a) "Primary" category components, (b) "Expansion" category components, and (c) "Existing Program" components

Issue:

The population of components in MRP-227-A includes "Primary," "Expansion," "Existing Program" and "No Additional Measures" category components, even though "No Additional Measures" components are not included as part of the sample of components that will be inspected in accordance with the MRP-227-A methodology. The "Scope of Program" program element for the PWR Reactor Internals Program does not include "No Additional Measures" components as part of the population of components that is included within the scope of the AMP. The methodology in MRP-227-A does not preclude the possibility that the same components identified as "No Additional Measures" components in MRP-227-A are ASME Section XI Examination Category B-N-2 or B-N-3 components for the STP units.

Request:

Justify the basis for omitting "No Additional Measures" components from the scope and population of components in the PWR Reactor Internals Program. Clarify whether any of the RVI "No Additional Measures" components at STP are defined as ASME Section XI Examination Category B-N-2 or B-N-3 components. If so, identify which "No Additional Measures" components are within the scope of the ASME Section XI Examination Category B-N-2 or B-N-3 requirements, and clarify whether the components will be inspected in accordance with the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD" Program (LRA AMP B2.1.1).

STPNOC Response:

License Renewal Application (LRA) Appendix B2.1.35 and License Renewal (LR) Basis Document Aging Management Program (AMP) Pressurized Water Reactor Internals (PWRI), Pressurized Water Reactor (PWR) Internals program, have been updated to identify a fourth group consisting of those PWR internals components for which the effects of all eight aging mechanisms are below the screening criteria were placed in the No Additional Measures group. No further action is required for managing the aging of the No Additional Measures components. The No Additional Measures components are not core support structures, and therefore are not covered by an AMP element such as the ASME Section XI Examination Category B-N-2 or B-N-3 requirements. These No Additional Measures components are show in the Aging Management Review (AMR) Table 3.1.1 Enclosure 2 of STP letter dated June 30, 2015 pages 33-35.

Enclosure 2 provides LRA Changes with Line-in/Line-out Annotations for B2.1.35

RAI 3.0.3.3.6-2 – Apparent component categorization inconsistencies

Background:

The background information in RAI 3.0.3.3.6-1 apply to this RAI. In addition, the "scope of program" program element in the PWR Reactor Internals Program identifies that the scope of the AMP includes the XL lower core plate as an "Expansion" category component for the AMP.

Issue:

In MRP-227-A, the EPRI MRP identifies XL lower core plates in Westinghouse-designed PWR as "Existing Program" components that are inspected in accordance with ASME Section XI Examination Category B-N-3 requirements and does not define these components as "Expansion" components. To be consistent with this protocol, the reactor vessel internals inspection plan (RVIIP), as submitted in the letter of June 30, 2015, identifies that the XL lower core plates are "Existing Program" components that will be inspected in accordance with ASME Section XI Examination Category B-N-3 requirements. Thus, there is an apparent inconsistency between the category identified for the XL lower core plates in the "Scope of Program" element and the category for these components identified in the RVIIP.

Request:

Clarify whether the XL lower core plates (one plate in each unit) are "Expansion" components or "Existing Program" components for the PWR Reactor Internals Program, or both. If the plates are "Expansion" components, identify and justify the basis for selecting the "Primary" components that are linked to the XL lower core plates as "Expansion" components for the AMP and the RVIIP.

STPNOC Response:

The XL lower core plates are Existing Program components for the PWR Reactor Internals Program.

LRA Appendix B2.1.35 and LR Basis Document AMP PWRI, PWR Reactor Internals program Scope of Program – Element 1 have been revised to show the XL lower core plate as an "Existing" category component.

Enclosure 2 provides LRA Changes with Line-in/Line-out Annotations for B2.1.35

RAI 3.0.3.3.6-3 – Response to A/LAI #1 – Lack of an MRP Letter 2013-025 Assessment

Background:

In the applicant's letter of June 30, 2015, the applicant provides its response to Applicant/Licensee Action Item (A/LAI) #1 on the MRP-227-A report, which requested the applicant to provide adequate demonstration that the assumptions for establishing the criteria in MRP-227-A are bounding the design of the RVI components at the applicant's facility. The EPRI MRP developed the criteria in EPRI MRP Letter No. 2013-025 to assist applicants of Westinghouse-designed PWRs in addressing the A/LAI request. In this letter the EPRI MRP recommended that applicants owning Westinghouse designed PWRs should provide their assessments of the following parameters:

- Demonstrate that the distance between the top of the active fuel and the upper core plate is greater than 12.2 inches
- Demonstrate that the average core power density is less than 124 watts/cm^3
- Demonstrate that the heat generation figure of merit, F, is less than or equal to 68 watts/cm^3

In the letter to the EPRI MRP, the staff agreed that demonstration of conformance with the acceptance criteria for these plant parameters would serve as a valid basis for concluding that the assumptions used in MRP-227-A are bounding for the design of the RVI at their facilities.

Issue:

The letter of June 30, 2015, does not include an assessment of the parameters listed above, as recommended in MRP Letter No. 2013-025.

Request:

Provide the basis why the response basis to A/LAI #1 in the letter of June 30, 2015, did not include an assessment of the three parameters listed above, as recommended in EPRI MRP Letter 2013-025. Justify why such an assessment would not be needed as part of the basis for concluding that the assumptions used to develop MRP-227-A are bounding for the design of the RVI components at STP Units 1 and 2.

STP Response:

The three parameters in RAI-3.0.3.3.6-3 and EPRI Letter MRP 2013-025, "MRP-227-A Applicability Template Guideline," October 14, 2013 [ML13322A454] are addressed for STP Unit 1 and Unit 2 within PWROG-14072-NP, Rev. 0, "South Texas Project Units 1 and 2 Summary Report for the Fuel Design/Fuel Management Assessments to Demonstrate MRP-227-A Applicability," June 3, 2015, and it is demonstrated that the parameters are met. A copy of this document has been provided in Enclosure 3 of this submittal.

The STP Unit 1 assessment concluded that the nominal distance between the top of the active fuel and bottom of the upper core plate, averaged over the first 19 fuel cycles of operation, was not less than 12.2 inches. STP Unit 1 has been operating at a rated power level of 3853 MWt for the last eight operating fuel cycles (Cycles 12 through 19). For the 193 fuel assembly core geometry of STP Unit 1, this corresponds to a core power density of 101.2 W/cm^3. It was also demonstrated that, considering the entire operating lifetime of the reactor, the average power density of the core shall be less than 124 W/cm^3 for a period of more than two effective full-power years. For the last eight operating fuel cycles, STP Unit 1 has kept its heat generation figure of merit range under 68 W/cm^3, and this range is representative of anticipated future operation.

The STP Unit 2 assessment concluded that the nominal distance between the top of the active fuel and bottom of the upper core plate, averaged over the first 17 fuel cycles of operation, was not less than 12.2 inches. STP Unit 2 has been operating at a rated power level of 3853 MWt for the last eight operating fuel cycles (Cycles 10 through 17). For the 193 fuel assembly core geometry of STP Unit 2, this corresponds to a core power density of 101.2 W/cm^3. It was also demonstrated that, considering the entire operating lifetime of the reactor, the average power density of the core shall be less than 124 W/cm^3 for a period of more than two effective full-power years. For the last eight operating fuel cycles, STP Unit 2 has kept its heat generation figure of merit range under 68 W/cm^3, and this range is representative of anticipated future operation.

Enclosure 3 contains a copy of PWROG-14072-NP, Rev. 0, "South Texas Project Units 1 and 2 Summary Report for the Fuel Design/Fuel Management Assessments to Demonstrate MRP-227-A Applicability," June 3, 2015.

RAI 3.0.3.3.6-4 – Response to A/LAI #2 – Comparison to UFSAR Information

Background:

The background information in RAI 3.0.3.3.6-1 apply to this RAI. In the applicant's response to A/LAI #2, the applicant states that the generic scoping and screening of the RVI, as summarized in the MRP-191 and MRP-232 reports (in order to support the inspection criteria in MRP-227-A), are applicable to STP Units 1 and 2 with no modifications for the components. The applicant states that the RVI components in the units are in conformance with the augmented inspection criteria in MRP-227-A for all components and that the protocols in MRP-227-A do not need to be modified under the criteria in A/LAI #2.

Issue:

In Section 4.1 of the updated final safety analysis report (UFSAR), the applicant identifies RVI design assembly or component modifications that have been or will be implemented in the units. Based on the UFSAR statements, the staff need to understand: (a) whether the specific RVI assemblies at STP include any design configurations that deviate from the RVI design assemblies and assembly components that were generically evaluated in the MRP-191, MRP-232, and MRP-227-A reports or were not evaluated in these reports, and (b) whether these deviations (if they exist) should have been more definitively assessed in the response that was provided to A/LAI #2. Apparent deviations for lower core support structure components are addressed in RAI 3.0.3.3.6-5.

Request:

Identify all RVI design assembly component configurations (other than those for the deviations on lower core support structure assembly components) that have not been evaluated by or differ from those generically evaluated in the MRP-191, MRP-232, and MRP-227-A reports, other than those for lower core support assembly components (which are the topic of RAI 3.0.3.3.6-5). For components that have corresponding components in the generic MRP evaluations but differ from the configurations in the generic evaluation, clarify how the stress levels and neutron fluences for these components compare to those assessed for corresponding components in the generic MRP design evaluations. Based on this comparison, justify why augmented inspection protocols for the components would not need to be proposed for the components on a plant-specific basis for the AMP. Similarly, for components not analyzed in the MRP reports, justify why plant-

specific aging management criteria would not need to be proposed for the components on a plant-specific basis for the AMP.

STPNOC Response:

The STP Unit 1 and Unit 2 reactor internals plant design was considered within the creation of MRP-191, *Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design (MRP-191)*. EPRI, Palo Alto, CA: 2006, as shown in Table 4-2 of MRP-191. Therefore, all RVI design assembly component configurations for STP Unit 1 and 2 were accounted for within MRP-191, Table 4-4, as concluded in PWROG Report, PWROG-15001-NP, Rev. 0, "South Texas Project Unit 1 and Unit 2 Summary Report for Applicant/Licensee Action Items 1, 2, and 7," June 2015. STP Unit 1 and 2 comply with A/LAI 2 of the NRC SE in MRP-227-A for all components. Therefore, STP Unit 1 and Unit 2 meet the requirement for application of MRP-227-A as a strategy for managing age-related material degradation in reactor internals components.

Enclosure 4 contains a copy of PWROG Report, PWROG-15001-NP, Rev. 0, "South Texas Project Unit 1 and Unit 2 Summary Report for Applicant/Licensee Action Items 1, 2, and 7," June 2015.

RAI 3.0.3.3.6-5 – Response to A/LAI #2 – Lower Core Support Assembly Deviations

Background:

The background information in RAI 3.0.3.3.6-4 apply to this RAI. Additionally, the "scope of program" program element for the PWR Reactor Internals Program and the tables of the Reactor Vessel Internals Inspection Plan (RVIIP) identify that the design of the RVI assemblies at STP do not include: (a) lower core support assemblies, (b) lower core support column bodies, or (c) lower core support column bolts. The lower core support column bodies and column bolts are defined as "Expansion" components in the MRP-227-A report.

Issue:

These deviations were not identified in the response to A/LAI #2 and change a number of generic "Primary" to "Expansion" category relationships for the RVIIP from those defined in the MRP-227-A report for these Westinghouse-designed internals.

Request:

Justify why the response to A/LAI #2 has not identified the lack of a lower core support structure assembly and lower core support column bodies and bolts (MRP-227-A "Expansion" components) as a deviations from the assessments in the MRP-191, MRP-232, and MRP-227-A reports. Clarify how these deviations would change the "Primary" to "Expansion" category relationships that need to be defined for the AMP and RVIIP when compared to those normally defined in the MRP-227-A report for Westinghouse-designed internals. Provide the basis why alternative "Expansion" component substitutions for these components would not need to be proposed for the AMP and RVIIP in order to be consistent with the total number of "Expansion" components defined in MRP-227-A for Westinghouse-designed internals.

STPNOC Response:

The STP Unit 1 and Unit 2 reactor internals plant designs were considered within the creation of MRP-191, *Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design (MRP-191).* EPRI, Palo Alto, CA: 2006, as shown in Table 4-2 of MRP-191. Therefore, the lower core support structure is accounted for by the XL lower core plate within MRP-191. There are no alternate component substitutions identified within MRP-227-A for STP Unit 1 and 2 and both units comply with A/LAI 2 of the NRC SE in MRP-227-A for all components. Therefore, STP Unit 1 and Unit 2 meet the requirement for application of MRP-227-A as a strategy for managing age-related material degradation in reactor internals components.

RAI 3.0.3.3.6-6 - Topic - Response to A/LAI #3 – Use of Inspection Data for CRGT Split Pins

Background:

In the applicant's letter of June 30, 2015, the applicant provides its response to A/LAI #3 on the MRP-227-A report, which requested that the applicant assess the need to replace or perform augmented inspections of their control rod guide tube (CRGT) support pins (split pins). The applicant's basis for resolving A/LAI #3 and for concluding that augmented inspections of the replaced CRGT split pins are not currently needed relies, in part, on the applicant's statement that data from industry inspections of replaced CRGT split pins made from Type 316 cold-worked stainless steel will be obtained from other U.S. (or foreign) licensees, the EPRI MRP, or other industry organizations and will be used to assess the need for developing augmented inspection criteria of the CRGT split pins at STP Units 1 and 2.

Issue:

1. The EPRI MRP has yet to identify in MRP-227-A or in the background reports for MRP-227-A that augmented inspections are part of the programmatic criteria for managing cracking or wear in replaced Westinghouse-design CRGT split pins made from Type 316 cold-worked stainless steel materials or that such data will be collected by the EPRI MRP for distribution to and evaluation by the industry licensee. Thus, some additional information is needed to clarify how the applicant will implement its process for collecting and assessing CRGT split pin inspection data in accordance with the PWR Reactor Internals Program.

2. If the CRGT splits pins are defined as ASME Section XI Examination Category B-N-3 removable core support structure components, the applicant will be required to inspect the components in accordance with their ISI program requirements for B-N-3 inspections, independent of the position taken in MRP-227-A for replaced split pins made from Type 316 cold-worked stainless steel materials.

Request:

1. Identify the plants that will be performing inspections of their replaced Type 316 cold-worked CRGT split pins which the applicant will use as the lead operating experience for managing aging in the CRGT split pins at STP Units 1 and 2. Identify the process or processes that will be used in accordance with the "Administrative Controls" or "Confirmation Process" elements of the PWR Reactor Internals Program to collect and compile the inspection data from these plants. Identify the criteria that will be implemented in accordance with the "monitoring and trending" program element of the AMP. Identify the

plant-specific "acceptance criteria" that will be used to assess such data and the "corrective actions" that will be taken if the acceptance criteria are not met.

2. Clarify whether the replaced CRGT split pins at STP are categorized as ASME Section XI Examination Category B-N-3 components (i.e. ASME removable core support structure components). If the split pins are defined as ASME removable core support structure components, justify why the components would not need to be inspected and managed for aging using either the "Existing Program" criteria in the PWR Reactor Internals Program (LRA B.2.1.35) or the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program (i.e., the ISI Program in LRA Section B2.1.1).

STPNOC Response:

STP follows operating experience (OE) and adjusts the program if needed, based on OE. The industry as a whole continues to share OE through the auspices of the MRP and PWROG. Any new found OE would be used by STPNOC to assess the need for developing augmented inspections for the split pins.

MRP-227-A, *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A).* EPRI, Palo Alto, CA: 2011, subsection 4.4.3, guidance for guide tube support pins (split pins) in Westinghouse plants is limited to plant-specific recommendations. The owner is directed to review and follow the original equipment manufacturer (OEM) recommendations for aging management and subsequent performance monitoring. Results of the detailed categorization and ranking of internals components contained in MRP-227-A, Table 3-3, identify only X-750 split pins as requiring specific actions to manage material aging in the period of extended operation (PEO); thus, no inspection or monitoring of the 316 stainless steel (SS) variant for control rod guide tube (CRGT) support pins is included in MRP-227-A, Table 4-9. As described In response to requests for additional information Set 28, dated June 30, 2015, RAI B2.1.35-1, STP Units 1 and 2 followed the OEM recommendation to replace the originally installed X-750 CRGT support pins with support pins fabricated from strain-hardened 316 SS material.

As listed in MRP-191, *Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design (MRP-191).* EPRI, Palo Alto, CA: 2006, Table 5-1, the 316 SS guide tube support pins were screened in for the aging degradation mechanisms of wear, fatigue, and irradiation stress relaxation/irradiation creep (ISR/IC). In MRP-232, *Materials Reliability Program: Aging Management Strategies for Westinghouse and Combustion Engineering PWR Internals (MRP-232, Revision 1).* EPRI, Palo Alto, CA: 2012, the 316 SS CRGT support pins were categorized as MRP-191 Category A, "no additional measures." The OEM recommendations do not require subsequent inspection of the 316 SS support pins. Long-term material behavior has been extensively studied from past testing and field experience; all identified degradation mechanisms, including irradiation-assisted stress corrosion cracking (IASCC), SCC, wear, fatigue, ISR/IC and embrittlement, have been assessed. Westinghouse Report, WCAP-15028-NP, Rev. 1, "Guide Tube Cold-Worked 316 Replacement Support Pin Development Program," June 24, 2011, concluded that the 316 SS CRGT support pins will perform all intended functions for the designated PEO with no requirement for post-installation inspections. STPNOC aging management programs comply with the OEM recommendations and MRP-227-A adequacy evaluation requirements for aging management of reactor internals CRGT 316 SS support pins.

The CRGT split pins are defined as ASME Section XI, Examination Category B-N-3 components at South Texas Project (STP) Units 1 and 2 as part of the upper internals assembly. MRP-227-A specifically states that there are no impacts to the plant-specific ASME Section XI program but that only specific items that may be part of the program were credited as part of the internals aging management inspection requirements [Inservice Inspection Program Plan for the South Texas Project Electric Generating Stations Units 1 & 2 Commercial Operation Dates Unit 1 August 25, 1988 Unit 2 June 19, 1989 Effective 10/08/2012]. There is no impact to the ASME Section XI inspection requirements as a result of MRP-227-A implementation at STP Units 1 and 2. STP Units 1 and 2 have therefore demonstrated that aging for split pins are adequately managed during the period of extended operation.

RAI 3.0.3.3.6-8 – Response to A/LAI #7 –Thermal Aging of CASS Upper Internals

Background:

In the applicant's letter of June 30, 2015, the applicant provides its response to A/LAI #7 on the MRP-227-A report, which addressed the issue of thermal aging embrittlement and neutron irradiation embrittlement in RVI components made from cast austenitic stainless steel (CASS).

Issue:

The response to A/LAI #7 uses the criteria in NRC License Renewal Issue 08-0030 (dated May 19, 2000) as the basis for concluding that thermal aging embrittlement will not be an aging management issue for RVI upper internals assembly support columns or column bases. Additional data is necessary to verify that thermal aging embrittlement will not be an aging mechanism of concern for these components during the period of extended operation.

Request:

Provide the plant-specific delta-ferrite contents for the CASS CF8 materials used to fabricate upper internals assembly support columns or column bases, and the equational criteria and plant specific chemistry alloy content data used to calculate the delta-ferrite contents of these components. As an alternative basis for resolving this issue (if applicable), the applicant may demonstrate that these components were appropriately evaluated in MRP-227-A or the background reports for MRP-227-A and were placed into FMECA Category A and "No Additional Measures" categories based on the conclusions that there are no consequences on RVI component intended functions if these components fail to maintain their structural integrity.

STPNOC Response:

With the exception of the upper internal support columns, CASS CF8 materials were used to fabricate the upper internal assembly as outlined in PWROG Report, PWROG-15001-NP, Rev. 0, "South Texas Project Unit 1 and Unit 2 Summary Report for Applicant/Licensee Action Items 1, 2, and 7," June 2015.

STP Unit 1 has fifty upper internals assembly –upper support columns, column bases that are comprised of CASS Grade CF8 material. Forty-eight of the fifty column bases are not susceptible to thermal embrittlement since they have a ferrite content less than or equal to 20 percent based on certified material test report (CMTR) data. Two of the fifty column bases have the potential to be susceptible to thermal embrittlement since CMTRs were not identified, and a conservative approach was taken. In MRP-191, *Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR*

Design (MRP-191). EPRI, Palo Alto, CA: 2006, Table 5-1, the upper support column, column bases CF8 are screened in for the material degradation effects of stress corrosion cracking (SCC), thermal embrittlement (TE) and irradiation embrittlement (IE). Taking into consideration all of the material degradation mechanisms, including IE, the upper support column base component was ranked in MRP-227-A as a "No Additional Measures" category component. Based on these results, the continued application of the MRP-227-A strategy for STP Unit 1 meets the requirement for managing age-related degradation of the STP Unit 1 CASS reactor vessel internals components.

STP Unit 2 has fifty upper internals assembly –upper support columns, column bases that are comprised of CASS Grade CF8 material. All fifty column bases are not susceptible to thermal embrittlement since they have a ferrite content less than or equal to 20 percent based on CMTR data. In MRP-191, *Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design (MRP-191).* EPRI, Palo Alto, CA: 2006, Table 5-1, the upper support column, column bases CF8 are screened in for the material degradation effects of SCC, TE, and IE. Taking into consideration all of the material degradation mechanisms, including IE, the upper support column base component was ranked in MRP-227-A as a "No Additional Measures" component. Based on these results, the continued application of the MRP-227-A strategy for STP Unit 2 meets the requirement for managing age-related degradation of the STP Unit 2 CASS reactor vessel internals components.

Enclosure 4 contains a copy of PWROG-15001-NP, Rev. 0, "South Texas Project Unit 1 and Unit 2 Summary Report for Applicant/Licensee Action Items 1, 2, and 7," June 2015.

3.0.3.3.6-9 – Response to A/LAI #8, Item 5 – RVI Environmentally-Assisted Fatigue

Background:

In the applicant's letter of June 30, 2015, the applicant provides its response to A/LAI #8, Item 5, on the MRP-227-A report, which addresses the bases that will be used to manage or adequately manage environmentally-assisted fatigue in PWR RVI components. In the applicant's response, the applicant identifies that the metal fatigue TLAAs have been included and evaluated in LRA Section 4.3.3 and that the PWR Reactor Internals Program will not be used as the basis for managing cracking induced by environmentally-assisted fatigue during the period of extended operation.

Issue:

Although the scope of LRA AMP B3.1 includes activities to monitor the impacts of environmentally-assisted fatigue on the CUF analyses for reactor coolant pressure boundary components, it is not evident whether similar activities will be applied to the CUF analyses for the RVI components listed in the background section of this RAI, and if so, how such activities will be applied to the cycle counting and CUF reanalysis criteria defined in the AMP.

Request:

Clarify whether the AMP's monitoring and trending activities for monitoring the impacts of environmental effects of the adequacy of components with CUF analyses are being extended to those RVI components with a CUF analysis. If not, identify the activities that will be performed to analyze or manage environmentally-assisted fatigue in the RVI components. Justify the response to this RAI.

STPNOC Response:

LRA AMP B3.1 has been revised to include activities to monitor the impacts of environmentally-assisted fatigue on the locations with fatigue usage calculations.

Enclosure 2 provides LRA Changes with Line-in/Line-out Annotations for LRA section B3.1.

Enclosure 4 provides LRA Changes with Line-in/Line-out Annotations for LRA Table A4.1 "STPNOC Regulatory Commitments"

RAI 3.0.3.3.6-10 – Adequacy of UFSAR Supplement Section A1.35

Background:

The current UFSAR supplement summary description for the PWR Reactor Vessel Internals Program is given in Section A1.35 of the LRA Appendix A. By letter dated June 30, 2015, the applicant submitted its updated version of the plant-specific PWR Reactor Internals Program (LRA Section B2.1.35) for staff review in order to respond to the staff's request in RAI B2.1.35-1. The updated version of the AMP provided in the June 30, 2015, letter updates the program element criteria for the AMP to be consistent with those provided in EPRI Technical Report No. 1022863, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guideline (MRP-227-A)," which was formally issued by the EPRI MRP in January 2012.

Issue:

In the letter of June 30, 2015, the applicant did not administratively update LRA Section A1.35, PWR Reactor Internals, to be consistent with the updated version of the PWR Reactor Internals Program (LRA Section B2.1.35) provided in the letter of June 20, 2015. Thus, the current version of LRA UFSAR Supplement Section A1.35, "PWR Vessel Internals," is out of date and must be updated to reflect the status of the AMP and reactor vessel internals inspection plan (RVIIP) that were submitted in the letter of June 30, 2015.

Request:

Justify why LRA Section A1.35 has not been updated to reflect that the current status of the AMP and RVIIP submitted in the letter of June 30, 2015. Specifically, justify why the USFAR supplement in Section A1.35 has not been updated to reflect the follow aspects of the program:

- Appropriate referenced ERPI Report for the AMP and UFSAR Supplement for the AMP is EPRI Technical Report No. 1022863, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guideline (MRP-227-A)"

- Protocols and activities for implementing the AMP and RVIIP in accordance with methodology in MRP-227-A are appropriately adjusted to account for deviations from the generic design and inspection and evaluation criteria in MRP-227-A or for the applicant's response bases for resolving specific Applicant/Licensee Action Items in the MRP-227-A report, as identified in the NRC safety evaluation for MRP-227-A dated December 16, 2011

- Population of components in the AMP include "Primary," "Expansion," "Existing Program," and "No Additional Measures" category components for the AMP

STPNOC Response:

STP letter dated February 27, 2012 provided a revised Appendix A Section A1.35 which updated the appropriate referenced ERPI Report for the AMP and UFSAR Supplement to EPRI Technical Report No. 1022863, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guideline (MRP-227-A)" and identified the protocols and activities for implementing the AMP and RVIIP in accordance with methodology in MRP-227-A.

LRA Appendix C outlines STPNOC's response to Applicant/Licensee Action Items.

Appendix B Section B2.1.35 outlines the population of components in the AMP include "Primary," "Expansion," "Existing Program," and "No Additional Measures" category components for the AMP.

RAI B2.1.13-5a – LR-ISG-2013-01 Inspection Frequency Followup

Background:

1. The applicant's response to RAI 3.0.3-2a Part (d) dated June 11, 2015, states the following, in part:

   When visual inspections detect any blistering, cracking, erosion, cavitation erosion, flaking, peeling, delamination, rusting and physical damage the coating is considered degraded. Degraded coatings are removed to sound material and replaced with new coating. The as-found degraded condition is documented in the corrective action program for trending. The NCS oversees the replacement of the degraded coatings assuring the extent of repaired or replaced coatings encompasses sound coating material. Review of STP's existing coating inspection program operating history demonstrates that the remediation of degraded coating conditions prior to returning the coating back in service is effective in managing the coating performance from one inspection to the next, with no change in inspection interval.

2. In regard to followup testing conducted to ensure that the extent of repaired or replaced coatings encompasses sound coating material, the response to RAI 3.0.3-2a Part (d) states that the nuclear coatings specialist's oversight of the replacement of the degraded coatings ensures that the extent of repaired or replaced coatings encompasses sound coating material.

3. Letters dated March 29, 2012, and May 10, 2012, state that the essential cooling water (ECW) pump internal coatings will be inspected on a nominal 10-year frequency. The May 10, 2012, letter states that ECW pumps are located upstream of self-cleaning strainers and the strainer size is sufficient to preclude tube blockage of downstream heat exchangers.

Issue:

1. LR-ISG-2013-01, "Aging Management of Loss of Coating or Lining Integrity for Internal Coatings/Linings on In-Scope Piping, Piping Components, Heat Exchangers, and Tanks," AMP XI.M42, "Internal Coatings/Linings for In-Scope Piping, Piping Components, Heat Exchangers, and Tanks," recommends that when peeling, delamination, blisters, or rusting are observed during inspections or when cracking and flaking that does not meet acceptance criteria is observed during inspections, the subsequent inspection interval is 4 years instead of 6 years. The responses to RAI 3.0.3-2a Part (d) state that the specific degraded coatings will be replaced and therefore inspections will continue at a 6-year interval. However, the 4 year inspection interval is recommended regardless of whether repairs are conducted on the degraded coatings detected during an inspection. With a known degradation mechanism potentially occurring in other locations with the same coating and environment, the staff concluded that subsequent inspections should be conducted more frequently than if no degradation was noted in prior inspections. The staff lacks sufficient information to conclude that a 6-year inspection interval is adequate when the extent of coating degradation, similar to the observed degradation that was repaired, is not known.

2. The "corrective actions" program element of LR-ISG-2013-01, "Aging Management of Loss of Coating or Lining Integrity for Internal Coatings/Linings on In-Scope Piping, Piping Components, Heat Exchangers, and Tanks," AMP XI.M42, "Internal Coatings/Linings for In-Scope Piping, Piping Components, Heat Exchangers, and Tanks," recommends that testing or examination be conducted to ensure that the extent of repaired coatings/linings encompasses sound material. The extent of blistering, peeling, and delamination is not typically detectable by visual inspection alone. The staff lacks sufficient information to conclude that follow-on testing or examination will be directed to be performed by the NCS.

3. Although the ECW pumps are located upstream of self-cleaning strainers, this in and of itself is not a sufficient basis to justify a nominal 10-year inspection frequency. The staff lacks sufficient information to conclude that the strainers will provide an effective barrier to flow blockage of downstream heat exchangers. Plant-specific operating experience of the ECW coatings has revealed degraded coatings.

Request:

1. With the exception of the internal coatings for the fire water storage tanks, state and justify the basis for how the extent of coatings that could be experiencing similar degradation to coated areas that were repaired will be determined in a reasonable time frame.

2. State whether testing and examination will be conducted during a coating repair to ensure that replaced coatings encompasses sound coating material.

3. In regard to the self-cleaning strainers downstream of the essential cooling water pumps, state:
    a. What backup indications are available to determine that fouling is not occurring on the self-cleaning strainers.
    b. Please provide the inspection interval of the strainer elements on the self-cleaning strainers.

STPNOC Response:

STP response was provided in Response to Request for Additional Information for the Review of the South Texas Project, Units 1 and 2, License Renewal Application – Set 32 (TAC Nos. ME4936 and ME4937), dated November 12, 2015, NOC-AE-15003303, as Supplemental Information - B2.1.13-5a, LR-ISG-2013-01 inspection frequency follow-up.

Enclosure 2

STPNOC LRA Changes with Line-in/Line-out Annotations

List of Revised LRA Sections

RAI	Affected LRA Section
3.0.3.3.6-1	B2.1.35
3.0.3.3.6-2	B2.1.35
3.0.3.3.6-9	B3.1

B2.1.35 PWR Reactor Internals

Program Description
The PWR Reactor Internals program manages cracking, loss of material, loss of fracture toughness, dimensional changes, and loss of preload for reactor vessel components that provide a core structural support intended function. The program implements the guidance of EPRI 1022863, *PWR Internals Inspection and Evaluation Guideline* (MRP-227-A, Rev .0) and EPRI 1016609, *Inspection Standard for PWR Internals* (MRP-228, Rev. 0). The program manages aging consistent with the inspection guidance for Westinghouse designated primary components in Table 4-3 of MRP-227-A and Westinghouse designated expansion components in Table 4-6 of MRP-227-A, and the Westinghouse designated existing components in Table 4-9 of MRP-227-A. Primary components are expected to show the leading indications of the degradation effects. The expansion components are specified to expand the primary component sample should the indications of the sample be more severe than anticipated. The aging effects of a third set of MRP-227-A internals locations are deemed to be adequately managed by existing program components whose aging is managed consistent with ASME Section XI Table IWB-2500-1, Examination Category B-N-3. A fourth group consisting of those PWR internals components for which the effects of all eight aging mechanisms are below the screening criteria were placed in the No Additional Measures group. No further action is required for managing the aging of the No Additional Measures components.

Program examination methods include visual examination (VT-3), enhanced visual examination (EVT-1), volumetric examination, and physical measurements. Bolting ultrasonic examination technical justifications in MRP-228 have demonstrated the indication detection capability to detect loss of integrity of PWR internals bolts, pins, and fasteners, such as baffle-former bolting. For some components, the MRP-227-A methodology specifies a focused visual (VT-3) examination, similar to the current ASME Code, Section XI, Examination Category B-N-3 examinations, in order to determine the general mechanical and structural condition of the internals by (a) verifying parameters, such as clearances, settings, and physical displacements; and (b) detecting discontinuities and imperfections, such as loss of integrity at bolted or welded connections, loose or missing parts, debris, corrosion, wear, or erosion. In some cases, VT-3 visual methods are used for the detection of surface cracking when the component material has been shown to be tolerant of easily detected large flaws. In some cases, where even more stringent examinations are required, enhanced visual (EVT-1) examinations or ultrasonic methods of volumetric inspection, are specified for certain selected components and locations.

The program provides both examination acceptance criteria for conditions detected as a result of monitoring the primary components, as well as criteria for expanding examinations to the expansion components when warranted by the level of degradation detected in the primary components. Based on the identified aging effect, and supplemental examinations if required, the disposition process results in an evaluation and determination of whether to accept the condition until the next examination or implement corrective actions. Any detected conditions that do not satisfy the examination acceptance criteria are required to be dispositioned through the corrective action program, which may require repair, replacement, or analytical evaluation for continued service until the next inspection.

The PWR Vessel Internals program is a new program that has been implemented. The program will include future industry operating experience, as it is incorporated into the future revisions of MRP-227-A, to provide reasonable assurance for long-term integrity of the reactor internals. The

reactor vessel internals included in the scope of the PWR Reactor Internals program are identified in Element 1. The scope of the program does not include welded attachments to the internal surface of the reactor vessel because these components are managed by the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD program (B2.1.1) (exam category B-N-2) and /or the Nickel-Alloy Aging Management Program (B2.1.34). The scope of the program also does not include BMI flux thimble tubes which are managed by the Flux Thimble Tube Inspection program (B2.1.21).

Aging Management Program Elements

The results of an evaluation of each element against the 10 elements described in Appendix A of NUREG-1800, *Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants* are provided below.

Scope of Program – Element 1

The scope of the program applies the guidance in MRP-227-A which provides augmented inspection and flaw evaluation methodology for assuring the functional integrity of Westinghouse reactor vessel internals. The scope of the PWR Reactor Internals program includes components that provide a core structural support intended function and are managed by the Westinghouse designated primary components in Table 4-3 of MRP-227-A and Westinghouse designated expansion components in Table 4-6 of MRP-227-A and applicable MRP-227-A methodology license renewal applicant action items. MRP-227-A Table 4-9 also identifies existing program components whose aging is managed consistent with ASME Section XI Table IWB-2500-1, Examination Category B-N-3.
Primary components are expected to show the leading indications of the degradation effects. The expansion components are specified to expand the primary component sample should the indications of the sample be more severe than anticipated. The aging effects of a third set of MRP-227-A internals locations are deemed to be adequately managed by existing program components whose aging is managed consistent with ASME Section XI Table IWB-2500-1, Examination Category B-N-3. <u>A fourth group consisting of those PWR internals components for which the effects of all eight aging mechanisms are below the screening criteria were placed in the No Additional Measures group. No further action is required for managing the aging of the No Additional Measures components.</u>

The STP reactor vessel internals are divided into the following major component groups: the lower core support assembly (including the entire core barrel assembly, baffle-former assembly, neutron shield panel, core support plate, and energy absorber assembly), the upper core support (UCS) assembly (including the upper support plate, support column, control rod guide tube assembly, upper core plate, and protective skirt), the incore instrumentation support structures (including the instrumentation columns (exit thermocouples), upper/lower tie plates, and instrumentation columns (BMI)), and miscellaneous alignment/interface components (including internals hold-down spring, upper core plate guide pins, and radial support keys including clevis inserts).

The following reactor vessel internals are included in the scope of the PWR Reactor Internals program:

1. Control rod guide tube assembly

- Guide plate (cards) [Primary component]

- Lower flange welds and adjacent base metal [Primary component]
- Guide Tube Support Pins [Existing programs component]

2. Core barrel assembly

- Upper core barrel flange weld and adjacent base metal [Primary component]
- Core barrel flange [Expansion component and Existing programs component]
- Core barrel vertical axial welds and adjacent base metal [Expansion component]
- Core barrel circumferential girth welds and adjacent base metal [Primary component]
- Core barrel outlet nozzle welds and adjacent base metal [Expansion component]
- Lower core barrel flange weld and adjacent base metal Addressed in AMR by Component Types of "RVI Core Barrel Assembly") [Primary component]

3. Baffle–former assembly

- Baffle–former bolting [Primary component]
- Baffle-former assembly baffle and former plates [Primary component]

4. Alignment and interfacing components

- Internals hold-down spring [Primary component]
- Clevis insert bolts [Existing programs component]
- Upper core plate alignment pins [Existing programs component]

5. Bottom Mounted Instrumentation (BMI) Column assembly

- BMI columns bodies [Expansion component]

6. Upper Internals assembly

- Upper core support skirt [Existing programs component]
- Upper Core Plate [Expansion component]

7. Lower internal assembly

- XL lower core-plate ~~[Expansion~~ Existing component]

The scope of the program also does not include welded attachments to the internal surface of the reactor vessel because these components are managed by the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD program (B2.1.1) (exam category B-N-2) and /or the Nickel-Alloy Aging Management Program (B2.1.34). The scope of the program also does not

include BMI flux thimble tubes which are managed by the Flux Thimble Tube Inspection program (B2.1.21).

The STP reactor vessel internals configuration does not include the lower internals assembly (lower support column bodies and lower core plate) noted in MRP-227-A.

The PWR Reactor Internals program is consistent with the following MRP-227-A assumptions (determination of applicability) which are based on PWR representative internals configurations and operational histories.

(1) STP has operated for less than 30 years of operation with high leakage core loading patterns. Operation with high leakage core loading was followed by implementation of a low-leakage fuel management pattern for the remaining operating life.

(2) STP operates at fixed power levels and does not usually vary power based on calendar or load demand schedule.

(3) STP has not implemented any design changes beyond those identified in industry guidance or recommended by Westinghouse.

Preventive Actions – Element 2

The PWR Reactor Internals program does not prevent degradation due to aging effects, but provides measures for monitoring to detect the degradation prior to loss of intended function. Preventive measures to mitigate aging effects such as loss of material and cracking include monitoring and maintaining reactor coolant water chemistry consistent with the guidelines of EPRI TR 1014986, *PWR Primary Water Chemistry Guidelines*, Volume 1. The primary water chemistry program is described separately in the Water Chemistry program (B2.1.2).

Parameters Monitored or Inspected – Element 3

The PWR Reactor Internals program monitors the following aging effects by inspection in accordance with the guidance of MRP-227-A or ASME Section XI Category B-N-3:

(1). Cracking
Cracking is due to stress corrosion cracking (SCC), primary water stress corrosion cracking (PWSCC), irradiation assisted stress corrosion cracking (IASCC), or fatigue /cyclical loading. Cracking is monitored with a visual inspection for evidence of surface breaking linear discontinuities or a volumetric examination. Surface examinations may also be used to supplement visual examinations for detection and sizing of surface-breaking discontinuities.

(2). Loss of Material
Loss of Material is due to wear. Loss of material is monitored with a visual inspection for gross or abnormal surface conditions.

(3). Loss of Fracture Toughness
Loss of fracture toughness is due to thermal aging or neutron irradiation embrittlement. The impact of loss of fracture toughness is indirectly monitored by using visual or volumetric examination techniques to monitor for cracking and by applying applicable reduced fracture toughness properties in the flaw evaluations if cracking is detected and is extensive enough to warrant a supplemental flaw growth or flaw tolerance evaluation.

(4). Dimensional Changes
Dimensional Changes are due to void swelling and irradiation growth, distortion or deflection. The program supplements visual inspection with physical measurements to monitor for any dimensional changes due to void swelling, irradiation growth, distortion, or deflection.

(5). Loss of Preload
Loss of preload is caused by thermal and irradiation-enhanced stress relaxation or creep. Loss of preload is monitored with a visual inspection for gross surface conditions that may be indicative of loosening in applicable bolted, fastened, keyed, or pinned connections.

The PWR Reactor Internals program manages the aging effects noted above consistent with the inspection guidance for Westinghouse designated primary components in Table 4-3 of MRP-227-A and Westinghouse designated expansion components in Table 4-6 of MRP-227-A. MRP-227-A also identifies Existing Program components whose aging is managed consistent with ASME Section XI Table IWB-2500-1, Examination Category B-N-3. See the component list in element 1 to identify Primary, Expansion, and Existing components. A fourth group consisting of those PWR internals components for which the effects of all eight aging mechanisms are below the screening criteria were placed in the No Additional Measures for aging management are specified.

Detection of Aging Effects – Element 4

The PWR Reactor Internals program detects aging effects through the implementation of the parameters monitored or inspected criteria and bases for Westinghouse designated Primary Components in Table 4-3 of MRP-227-A and for Westinghouse designated Expansion Components in Table 4-6 of MRP-227-A. The aging effects of a third set of MRP-227-A internals locations identified in Table 4-9 of MRP-227-A are deemed to be adequately managed by existing program components whose aging is managed consistent with ASME Section XI Table IWB-2500-1, Examination Category B-N-3.

One hundred percent of the accessible volume/area of each component will be examined for the Primary and Expansion components inspection category components. The minimum examination coverage for primary and expansion inspection categories is 75 percent of the component's total (accessible plus inaccessible) inspection area/volume be examined. When addressing a set of like components (e.g. bolting), the minimum examination coverage for primary and expansion inspection categories is 75 percent of the component's total population of like components (accessible plus inaccessible).

If defects are discovered during the examination, STP enters the information into the STP corrective action program and evaluates whether the results of the examination ensure that the component (or set of components) will continue to meet its intended function under all licensing basis conditions of operation until the next scheduled examination. Engineering evaluations that demonstrate the acceptability of a detected condition will be performed consistent with WCAP-17096-NP.

Monitoring and Trending – Element 5

The program provides both examination acceptance criteria (See Element 6) for conditions detected as a result of monitoring the primary components as described in Element 4, as well as criteria for expanding examinations to the expansion components when warranted by the level of degradation detected in the primary components. Based on the identified aging effect, and

supplemental examinations if required, the disposition process results in an evaluation and determination of whether to accept the condition until the next examination or implement corrective actions. Any detected conditions that do not satisfy the examination acceptance criteria (See Element 6) are required to be dispositioned through the corrective action program (See Element 7), which may require repair, replacement, or analytical evaluation for continued service until the next inspection.

Acceptance Criteria – Element 6

Examination acceptance for the Primary and Expansion component examinations are consistent with Section 5 of MRP-227-A. ASME Section XI section IWB-3500 acceptance criteria apply to Existing Programs components. The following examination acceptance criteria apply to the STP reactor vessel internals:

Visual examination (VT-3) and enhanced visual examination (EVT-1)

For existing program components, the ASME Code Section XI, Examination Category B-N-3 provides the following general relevant conditions for the visual (VT-3) examination of removable core support structures.

(1) Structural distortion or displacement of parts to the extent that component function may be impaired,

(2) Loose, missing, cracked, or fractured parts, bolting, or fasteners,

(3) Corrosion or erosion that reduces the nominal section thickness by more than 5 percent,

(4) Wear of mating surfaces that may lead to loss of function; and

(5) Structural degradation of interior attachments such that the original cross-sectional area is reduced more than 5 percent.

In addition, for the visual examinations (VT-3) of Primary and Expansion components, the PWR Reactor Internals program is consistent with the more specific descriptions of relevant conditions provided in Table 5-3 of MRP-227-A. EVT-1 examinations are used for detecting small surface breaking cracks and surface crack length sizing when used in conjunction with sizing aids. EVT-1 examination has been selected to be the appropriate NDE method for detection of cracking in plates or their welded joints. The relevant condition applied for EVT-1 examination is the same as found for cracking in ASME Section XI section 3500 which is crack-like surface breaking indications.

Volumetric examination

Individual bolts are accepted (pass/fail acceptance) based on the detection of relevant indications established as part of the examination technical justification. When a relevant indication is detected in the cross-sectional area of the bolt, it is assumed to be non-functional and the indication is recorded. Bolted assemblies are evaluated for acceptance based on meeting a specified number and distribution of functional bolts. Acceptance criteria for volumetric examination of STP reactor internals bolting are consistent with Table 5-3 of MRP-227-A.

Physical Measurements

Physical measurement of the internals hold down spring is not required because STP internals hold down spring are fabricated from 403 stainless steel.

Corrective Actions – Element 7

The following corrective actions are available for the disposition of detected conditions that exceed the examination acceptance criteria:

(1) Supplemental examinations to further characterize and potentially dispose of a detected condition consistent with Section 5.0 of MRP-227-A;

(2) Engineering evaluation that demonstrates the acceptability of a detected condition consistent with WCAP-17096-NP;

(3) Repair, in order to restore a component with a detected condition to acceptable status (ASME Section XI); or

(4) Replacement of a component with an unacceptable detected condition (ASME Section XI)

(5) Other alternative corrective action bases if previously approved or endorsed by the NRC.

Relevant indications failing to meet applicable acceptance criteria are repaired or replaced in accordance with plant procedures. Appropriate codes and standards are specified in both the "ASME Section XI Repair, Replacement, and Post-Maintenance Pressure Testing" procedure and in design drawings. Quality assurance requirements for repair and replacement activities are also included in the STP Operations Quality Assurance Plan.

STP site QA procedures, review and approval process, and administrative controls are implemented in accordance with the requirements of 10 CFR 50 Appendix B and are acceptable in addressing corrective actions. The QA program includes elements of corrective action, confirmation process and administrative controls, and is applicable to the safety-related and non-safety related systems, structures, and components that are subject to aging management review.

Confirmation Process – Element 8

STP site QA procedures, review and approval process, and administrative controls are implemented in accordance with the requirements of 10 CFR 50 Appendix B and are acceptable in addressing the confirmation process. The QA program includes elements of corrective action, confirmation process and administrative controls and is applicable to the safety-related and non-safety related systems, structures and components that are subject to aging management review.

Administrative Controls – Element 9

STP site QA procedures, review and approval process, and administrative controls are implemented in accordance with the requirements of 10 CFR 50 Appendix B and are acceptable in addressing administrative controls. The QA program includes elements of corrective action, confirmation process and administrative controls and is applicable to the safety-related and non-

safety related systems, structures and components that are subject to aging management review.

Operating Experience — Element 10

Relatively few incidents of PWR internals aging degradation have been reported in operating U.S. commercial PWR plants. However, a considerable amount of PWR internals aging degradation has been observed in European PWRs, with emphasis on cracking of baffle-former bolting. The experience reviewed includes NRC Information Notice 84-18, Stress Corrosion Cracking in PWR Systems and NRC Information Notice 98-11, Cracking of Reactor Vessel Internal Baffle Former Bolts in Foreign Plants. Most of the industry operating experience reviewed has involved cracking of austenitic stainless steel baffle-former bolts, or SCC of high-strength internals bolting. SCC of control rod guide tube split pins has also been reported.

Several other items with existing or suspected material degradation concerns that have been identified for PWR components are wear in thimble tubes and potentially in control guide cards and observed cracking in some high-strength bolting and in control rod guide tube alignment (split) pins. The latter are conditions that have been corrected primarily through bolt replacement with less susceptible material and improved control of pre-load.

Based on industry operating experience, STP replaced the Alloy-750 guide tube support pins (split pins) with strained hardened (cold worked) 316 stainless steel pins during Refueling Outage 1RE12 (Spring 2005) for Unit 1 and Refueling Outage 2RE11 (Fall 2005) for Unit 2. The replacement was conducted to reduce the susceptibility for stress corrosion cracking in the split pins. There were no cracked Alloy X-750 pins discovered during the replacement process.

The ASME Code, Section XI, Examination Category B-N-3 examinations of core support structures conducted during Refueling Outage 1RE15 (Fall 2009) for Unit 1, and Refueling Outage 2RE14 (Spring 2010) for Unit 2, did not identify any conditions that required repair, replacement or evaluation.

The ISI Program portion of the PWR Reactor Internals program at STP is updated to account for industry operating experience. ASME Section XI is also revised every three years and addenda issued in the interim, which allows the code to be updated to reflect operating experience. The requirement to update the ISI Program to reference more recent editions of ASME Section XI at the end of each inspection interval ensures the ISI Program reflects enhancements due to operating experience that have been incorporated into ASME Section XI.

With exception of the ASME Section ISI portions, the PWR Reactor Internals program will be a new program and has no direct programmatic history. A key element of the MRP-227-A program is the reporting of aging of reactor vessel components. STP, through its participation in PWR Owners Group and EPRI-MRP activities, will continue to benefit from the reporting of inspection information and will share its own operating experience with the industry through those groups or INPO, as appropriate.

As additional Industry and applicable plant-specific operating experience become available, the OE will be evaluated and appropriately incorporated into the program through the STP Corrective Action and Operating Experience Programs. This ongoing review of OE will continue throughout the period of extended operation, and the results will be maintained on site. This process will confirm the effectiveness of this new license renewal aging management program by incorporating applicable OE and performing self assessments of the program.

Conclusion

The implementation of the PWR Reactor Internals program provides reasonable assurance that aging effects will be adequately managed such that the systems and components within the scope of this program will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B3.1 METAL FATIGUE OF REACTOR COOLANT PRESSURE BOUNDARY

Program Description

The Metal Fatigue of Reactor Coolant Pressure Boundary program manages fatigue cracking caused by anticipated cyclic strains in metal components of the RCPB. The program ensures that actual plant experience remains bounded by the transients assumed in the design calculations, or that appropriate corrective actions maintain the design and licensing basis by other acceptable means.

The Metal Fatigue of Reactor Coolant Pressure Boundary program consists of cycle counting activities. The program will be enhanced to monitor and trend fatigue usage at selected locations in the reactor coolant pressure boundary and reactor vessel internals. The program will be enhanced to include additional transients and locations identified by the evaluation of ASME Section III fatigue analyses, locations necessary to ensure accurate calculations of fatigue, and the NUREG/CR-6260 locations for a newer-vintage Westinghouse Plant. The set includes fatigue monitoring of the NUREG/CR-6260 sample locations for a newer-vintage Westinghouse Plant, plant-specific bounding environmentally assisted fatigue (EAF) locations in the reactor coolant pressure boundary, and reactor vessel internals locations with fatigue usage calculations. The supporting environmental life correction factor calculations were performed with NUREG/CR-6583 for carbon and low alloy steels and with NUREG/CR-5704 for austenitic stainless steels.

The Metal Fatigue of Reactor Coolant Pressure Boundary program tracks the occurrences of selected transients and will be enhanced to monitor the cumulative usage factors (CUFs) at selected locations using one of the following methods:

1) The Cycle Counting (CC) method does not periodically calculate CUF; however, transient event cycles affecting the location (e.g. plant heatup and plant cooldown) are counted to ensure that the numbers of transient events assumed by the design calculations are not exceeded.

2) The Cycle Based Fatigue (CBF) management method utilizes the CC results and stress intensity ranges generated with the ASME III methods that use six stress-tensors to perform periodic CUF calculations, consistent with RIS 2008-30, *Fatigue Analysis of Nuclear Power Plant Components* for a selected location. The fatigue accumulation is tracked to determine approach to the ASME allowable fatigue limit of 1.0.

The Metal Fatigue of Reactor Coolant Pressure Boundary program continuously monitors plant data, and maintains a record of the data collected. The collected data are analyzed to identify operational transients and events, calculate usage factors for selected monitored locations, and compare the calculated usage factors to allowable limits. Periodic review of the calculations ensures that usage factors will not exceed the allowable value of 1.0 without an appropriate evaluation and any further necessary actions. If a cycle count or CUF value increases to a program action limit, corrective actions will be initiated to evaluate the design limits and determine appropriate specific corrective actions. Action limits permit completion of corrective actions before an assumed number of events in a fatigue analysis is exceeded.

NUREG-1801 Consistency

The Metal Fatigue of Reactor Coolant Pressure Boundary program is an existing program that, following enhancement, will be consistent with NUREG-1801, Section X.M1, Metal Fatigue of Reactor Coolant Pressure Boundary.

Exceptions to NUREG-1801

None

Enhancements

Prior to the period of extended operation, the following enhancements will be implemented in the following program elements:

Scope of Program (Element 1) and Monitoring and Trending (Element 5)

Procedures will be enhanced to include locations identified by the evaluation of ASME Section III fatigue analyses, locations necessary to ensure accurate calculations of fatigue, and the NUREG/CR-6260 locations for a newer-vintage Westinghouse Plant. The set includes fatigue monitoring of the NUREG/CR-6260 sample locations for a newer-vintage Westinghouse Plant, plant-specific bounding environmentally assisted fatigue (EAF) locations in the reactor coolant pressure boundary, and reactor vessel internals locations with fatigue usage calculations.

Scope of the Program (Element 1), and Parameters Monitored or Inspected (Element 3)

Procedures will be enhanced to include additional transients that contribute significantly to fatigue usage identified by the evaluation of ASME Section III fatigue analyses.

Scope of the Program (Element 1)

Procedures will be enhanced to ensure the fatigue crack growth analyses, which support the leak-before-break analyses and ASME Section XI evaluations, remain valid by counting the transients used in the analyses.

Detection of Aging Effects (Element 4)

The procedures will be enhanced to 1) include additional transients necessary to ensure accurate calculations of fatigue, 2) fatigue usage monitoring at specified locations, and 3) specify the frequency and process of periodic reviews of the results of the monitored cycle count and CUF data at least once per fuel cycle. This review will compare the results against the corrective action limits to determine any approach to action limits and any necessary revisions to the fatigue analyses will be included in the corrective actions.

Monitoring and Trending (Element 5)

STP will perform a review of design basis ASME Class 1 component fatigue evaluations to determine whether the NUREG/CR-6260-based components that have been evaluated for the effects of the reactor coolant environment and reactor vessel internals on fatigue usage are the limiting components for the STP configuration. If more limiting components are identified, the most limiting component will be evaluated for the effects of the reactor coolant environment on

fatigue usage. If the limiting location consists of nickel alloy, the methodology for nickel alloy in NUREG/CR-6909 will be used to perform the environmentally-assisted fatigue calculation.

Preventive Actions (Element 2) and Acceptance Criteria (Element 6)

The procedures will be enhanced to include additional cycle count and fatigue usage action limits, which will invoke appropriate corrective actions if a component approaches a cycle count action limit or a fatigue usage action limit. Action limits permit completion of corrective actions before the design limits are exceeded. The acceptance criteria associated with the NUREG/CR-6260 sample locations for a newer vintage Westinghouse plant will account for environmental effects on fatigue.

Cycle Count Action Limits:

Cycle count action limits are selected to initiate corrective action when the cycle count for any of the critical thermal or pressure transients is projected to reach the design limit within the next three fuel cycles.

CUF Action Limits:

CUF action limits require corrective action when the calculated CUF for any monitored location is projected to reach 1.0 within the next three fuel cycles.

Corrective Actions (Element 7)

Procedures will be enhanced to include appropriate corrective actions to be invoked if a component approaches a cycle count or CUF action limit.

If a cycle count action limit is reached, acceptable corrective actions include:

1) Review of fatigue usage calculations:

a) To identify the components and analyses affected by the transient in question.

b) To determine whether the transient in question contributes significantly to CUF.

c) To ensure that the analytical bases of the high energy line break (HELB) locations are maintained.

2) Evaluation of remaining margins on CUF.

3) Review of fatigue crack growth and stability analyses which support the leak before break exemptions and relief from the ASME Section XI flaw removal or inspection requirements to ensure that the analytical bases remain valid. Re-analysis of a fatigue crack growth analysis must be consistent with or reconciled to the originally submitted analysis and receive the same level of regulatory review as the original analysis.

4) Redefinition of the specified number of cycles (e.g., by reducing specified numbers of cycles for other transients and using the margin to increase the allowed number of cycles for the transient that is approaching its specified number of cycles).

5) Redefinition of the transient to remove conservatism in the pressure and temperature ranges.

These preliminary actions are designed to determine how close the approach is to the 1.0 limit, and from those determinations, set new action limits. If the CUF has approached 1.0 then further actions described below for cumulative fatigue usage action limits may be invoked.

If a CUF action limit is reached acceptable corrective actions include:

1) Repair the component.

2) Replace the component. If a limiting component is replaced, assess the effect on locations monitored by the program. If a limiting component is replaced, resetting its cumulative fatigue usage factor to zero, a component which was previously bounded by the replaced component will become the limiting component and may need to be monitored.

3) Perform a more rigorous analysis of the component to demonstrate that the design code limit will not be exceeded.

Operating Experience

The STP industry operating experience program reviews industry experience, including experience that may affect fatigue management, to ensure that applicable experience is evaluated and incorporated in plant analyses and procedures. Any necessary evaluations are conducted under the plant corrective action program.

The Metal Fatigue of Reactor Coolant Pressure Boundary program was implemented in response to industry experience that indicated that the design basis set of transients used for fatigue analyses of the reactor coolant pressure boundary did not include some significant transients, and therefore might not be limiting for components affected by them. Examples:

Thermal stratification of pressurizer surge line piping:

In response to NRC Bulletin 88-11, Westinghouse performed a plant-specific evaluation of STP pressurizer surge lines. The surge line stratification analysis was based on STP design transients. It was concluded that thermal stratification does not affect the integrity of the pressurizer surge lines. STP responses to NRC Bulletin 88-11 describe the inspections, analyses, and procedural revisions made to ensure that thermal stratification does not affect the integrity of the pressurizer surge lines. In addition, the responses noted that fatigue analyses were updated to ensure compliance with applicable codes and license commitments.

Thermal fatigue cracking in normally-isolated piping:

In 1988, as identified in NRC Bulletin 88-08, there were several instances of thermal fatigue cracking in normally stagnant lines attached to reactor coolant system (RCS) piping. This issue was addressed by utilities by conducting evaluations and monitoring to ensure that further leakage would not occur. STP performed a complete analysis of systems connected to the RCS. The review concluded that the potential for the described thermal conditions existed only in the normal charging, alternate charging, and auxiliary spray lines. However, these systems are separated and only hot water can leak through the charging and auxiliary spray lines, reducing the potential for thermal cycling.

Enclosure 3

Additional Supporting Documents

RAI	Additional Information
3.0.3.3.6-3	PWROG-14072-NP, Rev. 0

PWROG-14072-NP, Rev. 0, "South Texas Project Units 1 and 2 Summary Report for the Fuel Design / Fuel Management Assessments to Demonstrate MRP-227-A Applicability," June 3, 2015.

Enclosure 5

STPNOC Regulatory Commitment

Table A4-1 *License Renewal Commitments*

Item #	Commitment	LRA Section	Implementation Schedule
30	Enhance the Metal Fatigue of Reactor Coolant Pressure Boundary program procedures to: • include additional locations necessary to ensure accurate calculations of fatigue, • include additional transients that contribute significantly to fatigue usage, • include counting of the transients used in the fatigue crack growth analyses, which support the leak-before-break analyses and ASME Section XI evaluations to ensure the analyses remain valid, • include additional transients necessary to ensure accurate calculations of fatigue, fatigue usage monitoring at specified locations, and specify the frequency and process of periodic reviews of the results of the monitored cycle count and CUF data at least once per fuel cycle, • include additional cycle count and fatigue usage action limits, which will invoke appropriate corrective actions if a component approaches a cycle count action limit or a fatigue usage action limit. The acceptance criteria associated with the NUREG/CR-6260 sample locations for a newer vintage Westinghouse plant will account for environmental effects on <u>fatigue locations in the reactor coolant pressure boundary, and reactor vessel internals locations with fatigue usage calculations,</u> and • include appropriate corrective actions to be invoked if a component approaches a cycle count action limit or a fatigue usage action limit. Acceptable corrective actions include fatigue reanalysis, repair, or replacement. Re-analysis of a fatigue crack growth analysis must be consistent with or reconciled to the originally submitted analysis and receive the same level of regulatory review as the original analysis.	B3.1	Complete no later than six months prior to the period of extended operation Inspections to be complete no later than six months prior to the PEO or the end of the last refueling outage prior to the PEO, whichever occurs later. CR 10-23605