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NUCLEAR REGULATORY COMMISSION
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November 30, 2015

Mr. G. T. Powell, Site Vice President
Technical Support and Oversight
STP Nuclear Operating Company
P. O. Box 289
Wadsworth, TX 77483

SUBJECT: 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)

Dear Mr. Powell:

By letter dated October 25, 2010, STP Nuclear Operating Company submitted an application pursuant to Title 10 of the Code of Federal Regulations Part 54, to renew operating licenses NPF-76 and NPF-80 for South Texas Project, Unit 1 and 2, for review by the U.S. Nuclear Regulatory Commission (NRC or the staff). The staff is reviewing the information contained in the license renewal application and has identified, in the enclosure, areas where additional information is needed to complete the review.

These requests for additional information were discussed with Arden Aldridge, and a mutually agreeable date for the response is within 30 days from the date of this letter. If you have any questions, please contact me by telephone at 301- 415-3873 or e-mail John.Daily@nrc.gov.

Sincerely,

/RA/

John W. Daily, Senior Project Manager
Projects Branch 1
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket Nos. 50-498 and 50-499

Enclosure:
Requests for Additional Information

cc: Listserv

November 30, 2015

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SOUTH TEXAS PROJECT, UNITS 1 AND 2
LICENSE RENEWAL APPLICATION
REQUEST FOR ADDITIONAL INFORMATION SET 34

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).

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.

ENCLOSURE

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.

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 No. 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.

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.

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 also applies 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; they 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.

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).

RAI 3.0.3.3.6-7 – Response to A/LAI #5: Comparison to UFSAR Information

This RAI originally dealt with identifying acceptance criteria to be applied to physical measurements on the RVI hold-down springs for the units as discussed in MRP-227-A, A/LAI #5, as compared to some apparently conflicting information on hold-down spring design found in UFSAR Section 4.1. The staff and the applicant discussed the 2 types of hold-down springs in a conference call (held on October 15 and 27, 2015), the one for the RVI internals package as a whole (which is the subject of the A/LAI item) and the ones on the tops of the individual fuel assembly bundles (which are the subject of the information in UFSAR Section 4.1). Since UFSAR Section 4.1 discusses the hold-down springs for the fuel assemblies, this RAI is not necessary and is dropped.

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.

RAI 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.

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 following 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

RAI B2.1.35-11/B3.1-11 – This RAI was a duplicate request to RAI 3.0.3.6-9 and therefore is dropped.

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.

<u>Request</u>:

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.

Letter to G.T Powell from J. Daily Dated November 30, 2015

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