



UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
WASHINGTON, D.C. 20555-0001

November 15, 2011

Mr. D. W. Rencurrel, Sr. Vice President  
Technical Support and Oversight  
STP Nuclear Operating Company  
P.O. Box 289  
Wadsworth, TX 77483

SUBJECT: REQUESTS FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE  
SOUTH TEXAS PROJECT, UNITS 1 AND 2, LICENSE RENEWAL  
APPLICATION – AGING MANAGEMENT PROGRAM, SET 8  
(TAC NOS. ME4936 AND ME4937)

Dear Mr. Rencurrel:

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, Units 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 Michael Berg, 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 at 301-415-3873 or by e-mail at [john.daily@nrc.gov](mailto:john.daily@nrc.gov).

Sincerely,

A handwritten signature in cursive script that reads "John W. Daily".

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

Docket Nos. 50-498 and 50-499

Enclosure:  
As stated

cc w/encl: Listserv

SOUTH TEXAS PROJECT, UNITS 1 AND 2,  
REQUEST FOR ADDITIONAL INFORMATION  
AGING MANAGEMENT PROGRAM, SET 8  
(TAC NOS. ME4936 AND ME4937)

**STP Head Closure Studs Program - LRA B2.1.3 (003)**

**RAI B2.1.3-3a**

Background

By letter dated August 15, 2011, the staff issued request for additional information (RAI) B2.1.3-3, requesting that the applicant describe whether or not the aging management review (AMR) line items addressed in license renewal application (LRA) Table 3.1.2-1 to manage cracking and loss of material of reactor head closure stud bolting components include the closure studs, nuts, washers, bushings and flange threads. The applicant was also requested to revise the LRA and site documentation consistent with the applicant's response to the RAI.

In its response dated September 15, 2011, the applicant stated that the component type in LRA Table 3.1.2-1 for the "RV Closure Head Bolts" will be revised to "RV Closure Head Bolting Assemblies."

Issue

In its review, the staff noted that although the applicant stated that the component type in LRA Table 3.1.2-1 for the "RV Closure Head Bolts" will be revised to "RV Closure Head Bolting Assemblies," the applicant did not provide a specific revision made to LRA Table 3.1.2-1, which currently addresses only "RV closure head bolts," excluding the other reactor head closure bolting components.

Similarly, the applicant did not provide a specific revision made to the on-site document for component screening for this program, "South Texas Project License Renewal Component Summary Screening Report, Id No. RCVI, Reactor Vessel and Internals," Rev. 3, which addresses only "RV closure head bolts."

Request

Revise LRA Table 3.1.2-1, consistent with the program scope, including "RV closure head bolts," and the other reactor head closure bolting components.

In addition, revise the site document, "South Texas Project License Renewal Component Summary Screening Report, Id No. RCVI, Reactor Vessel and Internals," consistent with the program scope.

ENCLOSURE

**RAI B2.1.3-1a**Background

By letter dated August 15, 2011, the staff issued RAI B2.1.3-1, requesting that the applicant describe whether or not the measured yield strength levels of the reactor head closure stud bolting materials, which are used at the applicant's facility, exceed 150 ksi. The applicant was also requested to clarify whether or not the applicant will not use reactor head closure bolting materials with measured yield strength greater than 150 ksi. In addition, the staff requested that if the program does not have such assurance, the applicant justify the adequacy of the applicant's program to manage cracking due to stress-corrosion cracking (SCC) of the high-strength material.

As part of its response dated September 15, 2011, the applicant stated that the program will be enhanced to preclude the use of stud assembly material having a measured yield strength greater than or equal to 150 ksi, except for the installed and spare components currently on site. The applicant also stated that LRA Appendix B2.1.3 and LRA Basis Document XI.M3 (B2.1.3), Reactor Head Closure Studs, will be revised to preclude the use of replacement closure stud assemblies fabricated from material with a measured yield strength greater than or equal to 150 ksi.

Issue

The staff noted that the applicant has not yet revised the LRA or program basis document in accordance with its RAI response.

Request

Revise the LRA and program basis document to preclude the use of replacement closure bolting material with a yield strength level greater than or equal to 150 ksi, consistent with its RAI response.

**RAI B2.1.3-2a**Background

By letter dated August 15, 2011, the staff issued RAI B2.1.3-2, requesting that the applicant provide additional information regarding the applicant's engineering evaluation and continued use of the partially damaged stud insert of Unit 2 (April 2007), inspections of the reactor vessel head closure bolting components, and related operating experience such as leakage events.

Issue

By letter dated September 15, 2011, the applicant responded to RAI B2.1.3-2. In its response, the applicant did not provide information regarding inspections conducted to monitor any additional adverse change in the load bearing areas of the partially damaged stud insert. The staff finds that this information is needed for the staff's evaluation to confirm that neither additional reduction nor flaw initiation in the load bearing areas has occurred beyond the original damage. The applicant's site documentation conservatively estimates the original damage (rolling) of the stud insert as 5.14 in<sup>2</sup>, which is 17 % of the total load bearing surfaces of the stud insert lugs.

In its review of the applicant's response and related information, the staff noted that applicant's updated final safety analysis report Table 5.2-1, "Applicable Code Addenda for RCS Components," indicates that the reactor vessel head of STP, Unit 2, is constructed in accordance with the 1971 edition through the Summer of 1973 addenda of American Society of Mechanical Engineers (ASME) Code, Section III. The staff also noted that NB-3232.2, NB-3233, and NB-3234 of the 1971 edition of ASME Code, Section III, specify the requirements for the maximum stress for bolts in normal, upset and emergency conditions, respectively. These provisions of the ASME Code require that the maximum value of the service stress at the periphery of the bolt-cross section shall not exceed the three times the stress values of ASME Code Section III, Appendix I, Table I-1.3 (that is, not to exceed the three times design stress-intensity values,  $S_m$ , for bolting materials for Class 1 components).

The staff finds that the reduced load bearing surfaces of the partially damaged (rolled) stud insert increase the stress level applied to the lugs of the stud insert such that loss of material due to wear and cracking due to SCC may be facilitated. In contrast with these adverse effects on aging, the applicant's response to RAI B2.1.3-2 does not indicate whether or not the partially damaged stud insert complies with the aforementioned requirements of the ASME Code Section III for the maximum service stress limit.

#### Request

1. Clarify:
  - a. Whether or not inspections have been conducted to monitor any additional adverse change in the load bearing areas of the damaged stud insert since the partially damaged stud insert was placed in service after the applicant's engineering evaluation.
  - b. If subsequent inspections have been performed, provide the results of the inspections to confirm that neither additional reduction nor flaw initiation in the load bearing areas has occurred beyond the original damage addressed above.
2. If the applicant has not conducted a subsequent inspection of the partially damaged stud insert, provide information regarding the schedule and examination methods for the subsequent inspection to be conducted.
3. Describe the applicant's operating experience to clarify whether or not any other stud or stud insert has experienced damage similar to that of the partially rolled stud insert.
4. In view of the adverse effects of the damaged stud insert on aging due to the increased stress levels,
  - a. Provide information to confirm whether or not the partially damaged stud insert complies with the aforementioned requirements of ASME Code Section III, NB 3232.2, NB-3233, and NB-3234 for the maximum service stress limit in the normal, upset and emergence conditions.
  - b. As part of the response, describe the location of the maximum service stress.
  - c. In addition, provide information to clarify whether or not the maximum service stress of the damaged stud insert in faulted conditions does not exceed the three times the stress values of ASME Code Section III, Appendix I, Table I-1.3 in a consistent manner with the aforementioned ASME Code requirements.

Alternatively, justify why the maximum stress of the damaged stud insert in the faulted conditions are acceptable to adequately maintain the intended function of the reactor head closure bolting components.

## **Balance of Plant**

### **RAI SBPB-02-01**

#### Background

The response to RAI B2.1.18-1, dated September 15, 2011, stated that there are no piping or valves within systems included only for the criterion in 10 CFR 54.4(a)(2) that are managed by the Buried Piping and Tanks Inspection Program (BPTIP), and that the piping in LRA Table 3.3.2-27 that credited the BPTIP for aging management was removed from the scope of license renewal (LR). This was not considered in the LR drawings submitted with the LRA where piping was indicated as being in the scope of LR for 10 CFR 54.4(a)(2).

A review of revised LRA drawings LR-STP-CT-5S199F00020#1 and #2 identified several examples where piping attached to the safety relate auxiliary feedwater storage tanks (AFST) was previously identified as in scope of LR for 10CFR 54.4(a)(2) and is shown on the revised drawings as no longer in scope of LR. The drawings contain the following note for these lines:

LR Note 1: The tank penetrations are above the tank minimum required water level to support the tank's LR intended function, and the tank penetrations are not associated with tank venting. In addition, the tank nozzles have piping extensions that are securely braced within the concrete that surrounds the stainless steel AFST tank. The attached nonsafety-related (NSR) piping has not been included in-scope for structural integrity, based on the tank penetrations being above the tank minimum required water level and based on the nozzle penetration being analyzed as seismic equivalent anchors. Since the attached NSR piping does not have a structural integrity attached function, and also since it does not have spatial interactions with safety-related components, the piping is not within the scope of LR based on criterion 10 CFR 54.4(a)(2).

#### Issue

This appears to conflict with Section 2.1.2.2 of the LRA and Appendix F of NEI 95-10. The South Texas Project LRA states in Section 2.1.2.2:

Nonsafety-related structure system and component's (SSCs) that are directly connected to safety-related SSCs were included within the scope of LR to ensure structural integrity of the safety related SSC up to the first seismic anchor or equivalent anchor past the safety/nonsafety interface.

NEI 95-10, Appendix F – Industry Guidance on Revised 54.4(a)(2) Scoping Criterion (Non-Safety Affecting Safety) states:

#### 4. Non Safety SSCs Directly Connected to Safety-Related SSCs

For non-safety SSCs directly connected to safety-related SSCs (typically piping systems), the non-safety piping and supports, up to and including the first equivalent anchor beyond the safety/non-safety interface, are within the scope of LR per 54.4(a)(2).

Request

The applicant is requested to:

1. Provide the basis for compliance with LRA Section 2.1.2.2 and NEI 95-10 Appendix F for the nonsafety-related SSCs directly connected to safety-related SSC with respect to the piping attached to the AFST shown on the revised LRA drawings LR-STP-CT-5S199F00020#1 and #2.
2. Identify all instances (LRA drawing number and grid coordinates) where nonsafety-related SSCs attached to safety-related SSCs were included in the scope of LR in the original submittal of the LRA and have now been deleted from the scope of LR.

**RAI SBPB-02-02**Background

RAI 2.3.3.5-01, dated July 12, 2011, questioned why the floating seals in the reactor makeup water storage tanks were not in scope for LR. In the response dated August 9, 2011, the applicant stated that floating seals in the reactor makeup water storage tanks are within the scope of LR for non-safety affecting safety under 10 CFR 54.4(a)(2). The licensee further stated that the seals are short-lived components with the seals being replaced based on the oxygen levels in the makeup water.

Issue

The applicant's position that the seals are not subject to AMR because they are replaced based on oxygen levels in the makeup water is unacceptable in that it does not comply with Section 2.1.1 of NUREG 1800, "Standard Review Plan for Review of LRA for Nuclear Power Plants," which states in part:

"The SSCs subject to an AMR are those that perform an intended function, as described on 10 CFR 54.4 and meet two criteria:

1. They perform such functions without moving parts or without a change in configuration or properties, as set forth in 10 CFR 54.21(a)(1)(i), (denoted as "passive" components and structures in this Standard Review Plan), and, they are not subject to replacement based on a qualified life or specified time period, as set forth in 10 CFR 54.21(a)(1)(ii), (denoted as "long-lived" structures and components)."

Using the oxygen levels in the makeup water as a basis for replacing the seals is a "performance based" approach which is not the equivalent of "a qualified life or specified time period" called for in NUREG 1800 and 10 CFR 54.21(a)(1)(ii).

Request

The applicant is requested to revise their position on the seals being subject to an AMR or provide a basis for replacement that complies with 10 CFR 54.21(a)(1)(ii).

November 15, 2011

Mr. D. W. Rencurrel, Sr. Vice President  
Technical Support and Oversight  
STP Nuclear Operating Company  
P.O. Box 289  
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Sincerely,  
**/RA/**  
John W. Daily, Sr. Project Manager  
Projects Branch 1  
Division of License Renewal  
Office of Nuclear Reactor Regulation

Docket Nos. 50-498 and 50-499

Enclosure:  
As stated

cc w/encl: Listserv

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\*concurrence via e-mail

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DATE	11/3/2011	11/14/2011	11/15/2011	11/15/2011

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Letter to D. W. Rencurrel from John W. Daily dated November 15, 2011

**SUBJECT: REQUESTS FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE SOUTH TEXAS PROJECT, UNITS 1 AND 2, LICENSE RENEWAL APPLICATION – AGING MANAGEMENT PROGRAM, SET 8 (TAC NOS. ME4936 AND ME4937)**

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