Nuclear Operating Company

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

December 15, 2011
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U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2746

South Texas Project
Units 1 and 2
Docket Nos. STN 50-498, STN 50-499
Response to Requests for Additional Information for the
South Texas Project License Renewal Application
Aging Management Program, Set 8 (TAC Nos. ME4936 and ME4937)

References: 1. STPNOC Letter dated October 25, 2010, from G. T. Powell to NRC Document Control Desk, "License Renewal Application" (NOC-AE-10002607) (ML103010257)
2. NRC letter dated November 15, 2011, "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 ME 4937)" (ML11306A155)

By Reference 1, STP Nuclear Operating Company (STPNOC) submitted a License Renewal Application (LRA) for South Texas Project (STP) Units 1 and 2. By Reference 2, the NRC staff requests additional information for review of the STP LRA. STPNOC's response to the request for additional information is provided in Enclosure 1 to this letter. Changes to LRA pages described in Enclosure 1 are depicted in line-in/line-out pages provided in Enclosure 2.

Revised regulatory commitments are provided in Enclosure 3. There are no other regulatory commitments provided in this letter.

Should you have any questions regarding this letter, please contact either Arden Aldridge, STP License Renewal Project Lead, at (361) 972-8243 or Ken Taplett, STP License Renewal Project regulatory point-of-contact, at (361) 972-8416.

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

Executed on 12/15/2011
Date

D. W. Rencurrel
Senior Vice President,
Technical Support & Oversight

KJT
Enclosure: 1. STPNOC Response to Requests for Additional Information
2. STPNOC LRA Changes with Line-in/Line-out Annotations
3. STPNOC LRA Revised Regulatory Commitments

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Enclosure 1

STPNOC Response to Requests for Additional Information

STPNOC Response to Requests for Additional Information

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.

STPNOC Response:

LRA Table 3.1.2-1 component type "RV Closure Head Bolts" is revised to "RV Closure Head Bolting Assemblies" consistent with RAI response B2.1.3-3 (see STPNOC letter to NRC dated November 4, 2011 (ML11319A026)(NOC-AE-11002750)). The "South Texas Project Component Summary Screening Report, Id No. RCVI, Reactor Vessel and Internals," component type RV Closure Head Bolts, is revised to RV Closure Head Bolting Assemblies consistent with RAI response B2.1.3-3. See Report, Id No. RCVI, Reactor Vessel and Internals, Revision 4.

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.

STPNOC Response:

LRA Appendix B2.1.3 and Table A4-1, Item 38 preclude the use of replacement closure bolting material with a yield strength level greater than or equal to 150 ksi, consistent with the response to RAI B2.1.3-1 [see STPNOC letter to NRC dated November 4, 2011 (ML11319A026) (NOC-AE-11002750)]. Revised LRA Basis Document XI.M3 (B2.1.3), Reactor Head Closure Studs precludes the use of replacement closure bolting material with a yield strength level greater than or equal to 150 ksi, consistent with RAI response B2.1.3-1. See LRA Basis Document XI.M3 (B2.1.3), Revision 4.

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^2, 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 1-1.3 (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.

STPNOC Response:

1. The required 10-year ASME Section XI inspection of all reactor closure head bolting materials was performed during the Unit 2 Fall 2008 refueling outage (2RE13). The stud hole inserts were ultrasonically inspected, as allowed by relief request RR-ENG-2-5 (approved by NRC correspondence dated June 17, 1999). The UT inspection did not identify any flaws in stud hole insert #30. Because this was a UT inspection, there was no information regarding whether the bearing surface of the insert had been reduced beyond the original damage.

2. LRA Appendix B2.1.3, LRA Table A4-1 and LRA Basis Document XI.M3 (B2.1.3), Reactor Head Closure Studs program, are revised to perform a remote VT-1 of stud insert #30 concurrent with the volumetric examination once every 10 years to verify no additional loss of bearing surface area has occurred.

3. No other stud or stud hole insert has experienced damage similar to that of insert #30.

4. Response to be provided by January 12, 2012

The revised LRA Appendix B2.1.3 is provided in Enclosure 2. Revised LRA commitment #38 is provided in Enclosure 3.

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 related 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)92).

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.

STPNOC Response:

1. LRA Section 2.1.2.2 was revised to add a discussion regarding non-safety-related piping directly connected to the auxiliary feedwater storage tank (AFST) piping penetrations. This change was sent to the NRC in STPNOC letter dated June 16, 2011 (NOC-AE-11002681)(ML11172A096).

2. There are no other instances where non-safety-related structures, systems, and components (SSCs) attached to safety-related SSCs, included in the original scope of License Renewal, were removed from the scope of License Renewal.

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).

STPNOC Response:

LRA Appendix A1.22, B2.1.22, Table A4-1 and LRA Basis Document XI.M38, Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components, are revised to include the aging management of the floating seals in the reactor makeup water storage tanks (RMWST). The program includes inspections of the floating seals. The inspections look for thinning along the sides and at seams, seam adhesion, seal flotation, and condensation buildup on top of the seals. The inspections also manipulate the seals, inspecting for elastomer hardening and any other changes in mechanical properties. The first inspection is to be accomplished within five years prior to the period of extended operation with follow-up inspections every five years thereafter. LRA Table 2.3.3-5, Section 3.3.2.1.5 and Table 3.3.2-5 are revised to add aging management lines for the RMWST floating seal. LRA Table A4-1 is revised to reflect that the implementation of AMP B2.1.22, Inspection of Internal Surfaces in Miscellaneous Piping and

Ducting Components, will be implemented during the 5 year period prior to the period of extended operation to be consistent with the first inspection of the RMWST floating seals.

See Enclosure 2 for revisions to LRA Appendix A1.22, B2.1.22, LRA Table 2.3.3-5, Section 3.3.2.1.5 and Table 3.3.2-5.

See Enclosure 3 for revision to LRA Table A4-1 Commitment 17.

Enclosure 2

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

List of Revised LRA Sections

RAI	Affected LRA Section
RAI B2.1.3-2a	Appendix B2.1.3 Table A4-1 Item 38 (see Enclosure 3)
RAI SBPB-02-02	Appendix A1.22 Appendix B2.1.22 Table 2.3.3-5 Section 3.3.2.1.5 Table 3.3.2-5 Table A4-1 Item 17 (see Enclosure 3)

B2.1.3 Reactor Head Closure Studs

Program Description

The Reactor Head Closure Studs program manages cracking and loss of material by conducting ASME Section XI inspections of reactor vessel flange stud hole threads, reactor head closure studs, nuts, washers, and bushings. The program includes periodic visual, surface, and volumetric examinations of reactor vessel flange stud hole threads, reactor head closure studs, nuts, washers, and bushings and performs visual inspections of the reactor vessel flange closure during primary system leakage tests. The STP program implements ASME Section XI code, Subsection IWB, 2004 Edition. Reactor vessel flange stud hole threads, reactor head closure studs, nuts, washers, and bushings are identified in ASME Section XI Tables IWB-2500-1 and are within the scope of license renewal. The program implements recommendations in NUREG-1339 and NRC Regulatory Guide 1.65, *Material and Inspection for Reactor Vessel Closure Studs*, to address reactor head stud bolting degradation except for yield strength of existing bolting materials. STP uses lubricants on reactor head closure stud threads after reactor head closure stud, nut, and washer cleaning and examinations are complete. The lubricants are compatible with the stud material and operating environment and do not include MoS_2 which is a potential contributor to stress corrosion cracking.

In conformance with 10 CFR 50.55a(g)(4)(ii), the STP ISI Program is updated during each successive 120-month inspection interval to comply with the requirements of the latest edition of the Code specified twelve months before the start of the inspection interval. STP will use the ASME Code Edition consistent with the provisions of 10 CFR 50.55a during the period of extended operation.

Potential cracking and loss of material conditions in reactor vessel flange stud hole threads, reactor head closure studs, nuts, washers, and bushings are detected through visual, surface, or volumetric examinations in accordance with ASME Section XI requirements in STP procedures every ten years. A remote VT-1 of stud insert #30 is performed concurrent with the volumetric examination once every 10 years to verify no additional loss of bearing surface area. These inspections are conducted during refueling outages. Reactor vessel studs are removed from the reactor vessel flange each refueling outage. Studs, nuts, washers, and bushings are stored in protective racks after removal. Reactor vessel flange holes are plugged with water tight plugs during cavity flooding. These methods assure the holes, studs, nuts, washers, and bushings are protected from borated water during cavity flooding. Reactor vessel flange leakage is detected prior to reactor startup during reactor coolant system pressure testing each refueling outage. The STP program has proven to be effective in preventing and detecting potential aging effects of reactor vessel flange stud hole threads, closure studs, nuts, washers, and bushings.

NUREG-1801 Consistency

The Reactor Head Closure Studs program is an existing program that is consistent, with exception to NUREG-1801, Section XI.M3, Reactor Head Closure Studs.

Exceptions to NUREG-1801

Program Elements Affected:

Scope of Program (Element 1)

Regulatory Guide 1.65 states that the ultimate tensile strength of stud bolting material should not exceed 170 ksi. One closure head insert has a tensile strength of 174.5 ksi. STP credits inservice inspections that are within the scope of this AMP, which are implemented in accordance with the STP Inservice Inspection Program, Examination Category B-G-1 requirements, as the basis for managing cracking in these components. This is in accordance with the "parameters monitored or inspected" and "detection of aging effects" program elements in NUREG 1801, Section XI.M3. In addition, the studs, nuts and washers are coated with a lubricant which is compatible with the stud materials, and the studs, nuts, and washers are protected from exposure to boric acid by removing them and plugging the reactor vessel flange holes during cavity flooding. Replacement reactor head closure bolting obtained in the future (not currently installed or on site as spare parts) will be fabricated from material with an actual measured yield strength less than 150 ksi.

Corrective Actions (Element 7)

NUREG-1801, Section XI.M3 specifies the use of Regulatory Guide 1.65 requirements for closure stud and nut material. STP uses SA-540, Grade B-24 (as modified by Code Case 1605) stud material. The use of this material has been found acceptable to the NRC for this application within the limitations discussed in Regulatory Guide 1.85, *Materials Code Case Acceptability.*

Enhancements

Scope of Program (Element 1)

Procedures will be enhanced to preclude the future use of replacement closure stud assemblies fabricated from material with an actual measured yield strength greater than or equal to 150 ksi. The use of currently installed components and any spare components which are currently on site is allowed.

Detection of Aging Effects (Element 4)

Procedures will be enhanced to perform a remote VT-1 of stud insert #30 concurrent with the volumetric examination once every 10 years to verify no additional loss of bearing surface area.

Operating Experience

Review of plant-specific operating experience has not revealed any program adequacy issues with the Reactor Head Closure Studs program for reactor vessel closure studs, nuts, washers, bushings, and flange thread holes. No cases of cracking due to SCC or IGSCC have been identified with STP reactor vessel studs, nuts, washers, bushings, and flange stud holes.

Review of the Refueling Outage Inservice Inspection Summary Reports for Interval 2 indicates there were no repair/replacement items identified with reactor vessel closure studs, nuts, washers, bushings, or flange thread holes. None of the repair/replacement items indicate any

implementation issues with the STP ASME Section XI Program for reactor closure studs, nuts, washers, bushings, or flange thread holes.

The ISI 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.

Conclusion

The continued implementation of the Reactor Head Closure Studs program provides reasonable assurance that aging effects will be 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

A1.22 INSPECTION OF INTERNAL SURFACES IN MISCELLANEOUS PIPING AND DUCTING COMPONENTS

The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components program manages cracking, loss of material, and hardening and loss of strength of the internal surfaces of piping, piping components, ducting, tanks and other components that are not inspected by other aging management programs.

The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components program is a new program that uses the work control process for preventive maintenance and surveillance to conduct and document inspections. The program performs visual inspections to detect aging effects that could result in a loss of component intended function. Visual inspections of internal surfaces of plant components are performed by qualified personnel during the conduct of periodic maintenance, predictive maintenance, surveillance testing, and corrective maintenance. Supplemental inspections not performed concurrently with planned work activities will be performed. The locations and intervals for these supplemental inspections are based on assessments of the likelihood of significant degradation and on current industry and plant-specific operating experience.

Additionally, visual inspections will be augmented by physical manipulation of at least 10 percent of available surface area of elastomers within the scope of the program, when appropriate for the component configuration and material, to detect hardening and loss of strength of internal surfaces of elastomers. In cases where internal surfaces are not available for visual inspection, an internal visual inspection may be substituted with a volumetric examination. The program includes volumetric examination of the tank bottoms of the auxiliary feedwater storage tanks and the firewater storage tanks from inside the tanks within 5 years prior to entering the period of extended operation and whenever the tanks are drained to confirm the absence of loss of material due to corrosion. The program also includes volumetric evaluation to detect stress corrosion cracking of the internal surfaces of stainless steel components exposed to diesel exhaust. The program includes visual inspections of the floating seals in the reactor makeup water storage tanks.

The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components program will be implemented prior to the period of extended operation. Industry and plant-specific operating experience will be evaluated in the development and implementation of this program.

B2.1.22 Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components

Program Description

The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components program manages cracking, loss of material, and hardening and loss of strength of the internal surfaces of piping, piping components, ducting, tanks and other components that are not inspected by other aging management programs.

The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components program is a new program that uses the work control process for preventive maintenance and surveillance to conduct and document inspections. The program performs visual inspections to detect aging effects that could result in a loss of component intended function. Visual inspections of internal surfaces of plant components are performed by qualified personnel during periodic maintenance, predictive maintenance, surveillance testing and corrective maintenance. Supplemental inspections not performed concurrently with planned work activities will be performed. The locations and intervals for these supplemental inspections are based on assessments of the likelihood of significant degradation and on current industry and plant-specific operating experience.

Additionally, visual inspections will be augmented by physical manipulation of at least 10 percent of available surface area of elastomers within the scope of the program, when appropriate for the component configuration and material, to detect hardening and loss of strength of internal surfaces of elastomers. In cases where internal surfaces are not available for visual inspection, an internal visual inspection may be substituted with a volumetric examination. The program includes volumetric examination of the tank bottoms of the auxiliary feedwater storage tanks and the firewater storage tanks from inside the tanks within 5 years prior to entering the period of extended operation and whenever the tanks are drained to confirm the absence of loss of material due to corrosion. The program also includes volumetric evaluation (ultrasonic examination) to detect stress corrosion cracking of the internal surfaces of stainless steel components exposed to diesel exhaust. The program includes visual inspections of the floating seals in the reactor makeup water storage tanks. The first inspection is to be accomplished within five years prior to the period of extended operation with follow-up inspections every five years thereafter.

This program will be initiated prior to entering the period of extended operation and provides for periodic inspection of a selected set of sample components within the scope of this program. The internal surfaces inspections are normally performed through scheduled preventive maintenance and surveillance inspections such that work opportunities are sufficient to detect aging and provide reasonable assurance that intended functions are maintained. Supplemental inspections not performed concurrently with planned work activities will be performed. The locations and intervals for these supplemental inspections will be based on assessments of the likelihood of significant degradation and on current industry and plant-specific operating experience.

NUREG-1801 Consistency

The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components program is a new program that, when implemented, will be consistent with exception to NUREG-1801, Section XI.M38, Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components.

Exceptions to NUREG-1801

Program Elements Affected:

Scope of Program (Element 1), Parameters Monitored or Inspected (Element 3), Detection of Aging Effects (Element 4), and Monitoring and Trending (Element 5)

NUREG-1801 Section XI.M38 provides for a program of visual inspections of the internal surfaces of miscellaneous steel piping and ducting components to ensure that existing environmental conditions are not causing material degradation that could result in a loss of component intended functions. The exceptions to NUREG-1801, Section XI.M38 are an increase to the scope of the materials inspected to include stainless steel, aluminum, copper alloy, stainless steel-cast austenitic, nickel alloys, glass and elastomers, in addition to steel, and an increase to the scope of aging effects to include hardening and loss of strength for elastomers. Additionally, visual inspections will be augmented (1) by physical manipulation of at least 10 percent of available surface area of elastomers within the scope of the program to detect hardening and loss of strength of elastomers when appropriate for the component configuration and material, (2) volumetric examinations of the tank bottoms of the auxiliary feedwater storage tanks and the firewater storage tanks from inside the tanks, to confirm the absence of loss of material due to corrosion, and (3) volumetric evaluation to detect stress corrosion cracking of the internal surfaces of stainless steel components exposed to diesel exhaust.

Enhancements

None

Operating Experience

The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components program is a new program; therefore, plant-specific operating experience to verify the effectiveness of the program is not available. However, visual inspections were conducted during periodic maintenance, predictive maintenance, surveillance testing and corrective maintenance. These records provided evidence of STP using maintenance opportunities to conduct internal inspections during normal plant activities. Industry operating experience that forms the basis for this program is included in the operating experience element of the corresponding NUREG-1801 aging management program. A review of plant condition reporting documents, as well as other STP current licensing basis documents, since 1998, was performed to ensure that there is no unique, plant-specific experience in addition to that in NUREG-1801. The review identified no unique operating experience.

Many of the plant condition reporting documents discussed above concerned corrosion found in HVAC systems. The corrective actions for these conditions generally included removal of the corrosion and painting to prevent recurrence.

As additional industry and plant-specific applicable operating experience becomes available, it will be evaluated and incorporated into the program through the STP condition reporting and operating experience programs.

Conclusion

The implementation of the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components program will provide reasonable assurance that aging effects will be 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.

Table 2.3.3-5 Reactor Makeup Water System

Component Type	Intended Function
Closure Bolting	Leakage Boundary (spatial) Pressure Boundary Structural Integrity (attached)
Flow Element	Pressure Boundary
Orifice	Leakage Boundary (spatial) Pressure Boundary, Throttle
Piping	Leakage Boundary (spatial) Pressure Boundary Structural Integrity (attached)
Pump	Pressure Boundary
Seal	Pressure Boundary
Tank	Pressure Boundary
Tubing	Leakage Boundary (spatial) Pressure Boundary
Valve	Leakage Boundary (spatial) Pressure Boundary Structural Integrity (attached)

3.3.2.1.5 Reactor Makeup Water System

Materials

The materials of construction for the reactor makeup water system component types are:

- Elastomer
- Stainless Steel
- Stainless Steel Cast Austenitic

Environment

The reactor makeup water system components are exposed to the following environments:

- Demineralized Water
- Plant Indoor Air

Aging Effects Requiring Management

The following reactor makeup water system aging effects require management:

- Hardening and loss of strength
- Loss of material
- Loss of preload

Aging Management Programs

The following aging management programs manage the aging effects for the reactor makeup water system component types:

- Bolting Integrity (B2.1.7)
- Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (B2.1.22)
- One-Time Inspection (B2.1.16)
- Water Chemistry (B2.1.2)

Table 3.3.2-5 Auxiliary Systems – Summary of Aging Management Evaluation – Reactor Makeup Water System

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Vol. 2 Item	Table 1 Item	Notes
Pump	PB	Stainless Steel Cast Austenitic	Plant Indoor Air (Ext)	None	None	VIII.I-10	3.4.1.41	A
Seal	PB	Elastomer	Demineralized Water (Int)	Hardening and loss of strength	Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (B2.1.22)	None	None	G, 2
Seal	PB	Elastomer	Plant Indoor Air (Ext)	Hardening and loss of strength	Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (B2.1.22)	VII.F2-7	3.3.1.11	E
Tank	PB	Stainless Steel	Demineralized Water (Int)	Loss of material	Water Chemistry (B2.1.2) and One-Time Inspection (B2.1.16)	VIII.E-29	3.4.1.16	A

Notes for Table 3.3.2-5:

Standard Notes:

A Consistent with NUREG-1801 item for component, material, environment, and aging effect. AMP is consistent with NUREG-1801 AMP.

E Consistent with NUREG-1801 for material, environment, and aging effect, but a different aging management program is credited or NUREG-1801 identifies a plant-specific aging management program.

G Environment not in NUREG-1801 for this component and material.

H Aging effect not in NUREG-1801 for this component, material, and environment combination.

Plant Specific Notes:

1 Loss of preload is conservatively considered to be applicable for all closure bolting.

2 The Reactor Make-up Water Storage Tank floating seal cannot be readily inspected from the treated water side. Credit for aging management of the floating seal is by physical inspection from the accessible topside of the floating seal following the inspection attributes in XI.M38, Inspection of Internal Surfaces. The XI.M38 inspections will determine if the elastomeric floating seal is experiencing hardening, and loss of strength.

Enclosure 3

STPNOC LRA Revised Regulatory Commitments

A4 LICENSE RENEWAL COMMITMENTS

Table A4-1 identifies proposed actions committed to by STPNOC for STP Units 1 and 2 in its License Renewal Application. These and other actions are proposed regulatory commitments. This list will be revised, as necessary, in subsequent amendments to reflect changes resulting from NRC questions and STPNOC responses. STPNOC will utilize the STP commitment tracking system to track regulatory commitments. The Condition Report (CR) number in the Implementation Schedule column of the table is for STPNOC tracking purposes and is not part of the amended LRA.

Table A4-1 License Renewal Commitments

Item #	Commitment	LRA Section	Implementation Schedule
38	Enhance the Reactor Head Closure Studs program procedures to: • preclude the future use of replacement closure stud assemblies fabricated from material with an actual measure yield strength greater than or equal to 150 ksi. The use of currently installed components and any spare components which are currently on site is allowed, and • perform a remote VT-1 of stud insert #30 concurrent with the volumetric examination once every 10 years to verify no additional loss of bearing surface area.	B2.1.3	Prior to the period of extended operation CR 11-22923-1
17	Implement the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components program as described in LRA Section B2.1.22.	B2.1.22	~~Prior to~~ During the five year period prior to the period of extended operation. CR 10-23274