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
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

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 13 (Supplemental) and Set 15 (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 February 15, 2012, "Requests for Additional Information for the Review of the South Texas Project, Units 1 and 2, License Renewal Application – Aging Management, Set 13 (TAC Nos. ME4936 and ME 4937)" (ML12039A240)
3. STPNOC letter dated March 12, 2012, from G. T. Powell to NRC Document Control Desk, "Response to Requests for Additional Information for the South Texas Project License Renewal Application – Aging Management Program, Set 13 (TAC Nos. ME4936 and ME4937)" (NOC-AE-12002802) (ML12079A015)
4. NRC letter dated March 21, 2012, "Requests for Additional Information for the Review of the South Texas Project, Units 1 and 2 License Renewal Application – Aging Management, Set 15 (TAC Nos. ME4936 and ME 4937)"(ML12065A201)
5. STPNOC letter dated March 29, 2012, from D. W. Rencurrel to NRC Document Control Desk, "Supplemental Response to Requests for Additional Information for the South Texas Project License Renewal Application – Aging Management Program, Set 13 and Set 14 (TAC Nos. ME4936 and ME4937)" (NOC-AE-12002825)

By Reference 1, STP Nuclear Operating Company (STPNOC) submitted a License Renewal Application (LRA) for South Texas Project (STP) Units 1 and 2. By References 2 and 4, the NRC staff requested additional information for review of the STP LRA. By References 3 and 5, STPNOC provided responses to the requests for additional information in Reference 2. Additional responses to Reference 2 and responses to Reference 4 are in Enclosure 1 to this letter. Changes to LRA pages described in Enclosure 1 are depicted as line-in/line-out pages in Enclosure 2.

One revised commitment and one new regulatory commitment are contained in Table A4-1 in Enclosure 3 to this letter. There are no other regulatory commitments in this letter.

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NRR

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 4/17/2012
Date

David W Rencurrel
D. W. Rencurrel
Chief Nuclear Officer

KJT

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

cc:

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Enclosure 1

STPNOC Response to Requests for Additional Information

STPNOC Supplemental Response to Requests for Additional Information

SOUTH TEXAS PROJECT, UNITS 1 AND 2
REQUEST FOR ADDITIONAL INFORMATION -
AGING MANAGEMENT, SET 13
(TAC NOS. ME4936 AND ME4937)

RAI 4.2.2-1 (059)

Background:

The STP Units 1 and 2 licensing renewal application (LRA) Tables 4.2-2, 4.2-3, 4.2-4 and 4.2-5, include data for the extended beltline materials (i.e., the inlet and outlet nozzles, nozzle shell, bottom head Taurus, bottom head dome, and associated welds). The NRC's Reactor Vessel Integrity Database (RVID) does not contain information for these materials.

Issue:

Licensees for all light water nuclear power reactors must meet fracture toughness requirements and maintain a material surveillance program for the reactor coolant pressure boundary. These requirements are set forth in Appendices G and H to 10 CFR Part 50. It has been demonstrated that some reactor pressure vessel (RPV) integrity evaluations are very sensitive to the consideration of new data, therefore information regarding RPV beltline materials should be consistent with the requirements in Generic Letter 92-01, and changes to values related to fracture toughness must provide a technical basis for the change.

Request:

1) Discuss the procedures used to determine the chemistry data, RT_{NDT}, margins and initial USE values for the extended beltline materials to demonstrate that you have applied consistent approaches in determining the above mentioned material information for all of the extended beltline materials. Although nozzle materials are listed in Tables 4.2-2 through 4.2-5, the nozzle-to-reactor pressure vessel (RPV) welds have not been included. If nozzle-to-RPV welds have predicted neutron fluence values greater or equal to 1×10^{17} n/cm^2 (E > 1MeV), add the data for those nozzle-to-RPV welds to Tables 4.2-2 though 4.2-5; include descriptions of the procedures used to determine the chemistry data, initial RT_{NDT}, margins and USE values.

2) Resolve the following discrepancies:

   a) For STP Unit 2 welds with heat number 90209, the RVID contains a nickel chemical composition value of 0.126 weight-percent. In the submittal, the nickel chemical composition value for this heat of material is 0.11

weight-percent. Explain the basis for the change in nickel chemical composition values for heat number 90209.

b) For STP Unit 1, the copper values for the upper to intermediate circumferential weld and the lower shell to lower head circumferential weld are not consistent in Tables 4.2-2 and 4.2-4 (0.1 and 0.35 weight percent copper respectively). Explain the differences and provide updated Tables with consistent values, as appropriate.

c) For STP Unit 2, the copper values for the upper to intermediate circumferential weld and lower shell to lower head circumferential weld are not consistent in Tables 4.2-3 and 4.2-5 (0.1 and 0.35 weight percent copper respectively). Explain the differences and provide updated Tables with consistent values, as appropriate.

STPNOC Response:

1) The procedure used to determine the material chemistry data listed in LRA Tables 4.2-2, 4.2-3, 4.2-4 and 4.2-5 is as follows:

   a) The copper (Cu) and nickel (Ni) values provided are from South Texas Project Updated Final Safety Analysis Report (UFSAR) Table 5.3-3 and Table 5.3-4.

   b) Where the Cu value for base metal components is not listed in UFSAR Table 5.3-3 and Table 5.3-4, a Cu value of 0.35% is assumed based on 10 CFR 50.61(c)(1)(iv)(A).

   c) The specific heat number recorded Cu and Ni values from the weld certification records are used for lower head torus longitudinal welds.

   d) Where the Cu value for welds is not listed in UFSAR Table 5.3-3 and Table 5.3-4 or the weld certification records, the Cu value used (0.1%) is the maximum allowable copper content listed in the STP reactor vessel specification.

   e) Where the Ni value for welds is not listed in the UFSAR or the weld certification records, a Ni value of 1.0% is assumed based on 10 CFR 50.61(c)(1)(iv)(A).

The procedure used to determine the initial reference temperature (RT(NDT)) data listed in LRA Tables 4.2-2 and 4.2-3 is as follows:

   a) The initial RT(NDT) values provided are from UFSAR Table 5.3-3 and Table 5.3-4.

   b) The specific heat number recorded initial RT(NDT) value from the weld certification records is used for lower head torus longitudinal welds.

c) Where the initial RT(NDT) values for welds are not listed in UFSAR Table 5.3-3 and Table 5.3-4 or the weld certification records, the initial weld RT(NDT) is set to the limiting value (-56°F) from 10 CFR 50.61(c)(1)(ii) for welds made with Linde 0091 and Linde 124 weld fluxes

The procedure used to determine normal to the principal working direction (NPWD) (ft-lb) data listed in Charpy Upper-Shelf Energy (Cv USE) LRA Tables 4.2-4 and 4.2-5 is as follows:

a) The NPWD values provided are from UFSAR Table 5.3-3 and Table 5.3-4.

b) The specific heat number recorded NPWD value from the weld certification records is used for lower head torus longitudinal welds initial RT(NDT). The initial USE information from UFSAR Table 5.3-3 and Table 5.3-4 is supplemented with measured values recorded in weld certification records for E-8018 heat numbers used in the STP Units 1 and 2 bottom head torus longitudinal weld seams.

c) Where the NPWD value for weld material is not listed in UFSAR Table 5.3-3 and Table 5.3-4 or the weld certification records, the generic USE value from CEN-622-A Final Report, "Generic Upper Shelf Values for Linde 1092, 124 and 0091 Reactor Vessel Welds, CEOG Task 839" December 1996, are used. CEN-622-A presents a review of weld flux types that were commonly used during the reactor vessel fabrication process by Combustion Engineering. Per CEN-622-A, generic initial USE values of 101 ft-lbs and 84 ft-lbs can be used for Linde 0091 and Linde 124 flux type welds, respectively.

The procedure used to determine margin term is defined by 10 CFR 50.61(c)(1)(iii), equation 2 ($M=2\sqrt{(\sigma(I)2+\sigma(\Delta)2)}$), where $\sigma(I)$ is the standard deviation associated with the initial measurement of RT(NDT) and $\sigma(\Delta)$ is the standard deviation associated with the prediction of ΔRT(NDT). The $\sigma(I)$ is set equal to 0°F for measured RT(NDT), and 17°F for limiting or generic values of RT(NDT), Cu, and Ni. Per 10 CFR 50.61(c)(1)(iii)(B), $\sigma(\Delta)$ is set equal one-half the value of ΔRT(NDT) with a maximum value of 28°F for welds and 17°F for base metal.

LRA Tables 4.2-2, 4.2-3, 4.2-4 and 4.2-5 are revised to add the extended beltline welds (i.e., the inlet and outlet nozzles, nozzle (upper) shell, lower head torus, and lower head torus to dome welds). Additionally, LRA Sections 4.2.2 and 4.2.3 are revised to discuss the fluence projection for these welds

LRA Tables 4.2-2 and 4.2-3 are revised to add the flux type for the nozzle (upper) to shell to intermediate shell circumferential welds and lower shell to lower head torus circumferential welds.

LRA Tables 4.2-4 and 4.2-5 Cu and NPWD values are revised based on the flux type for the lower shell to lower head torus circumferential welds and nozzle (upper) shell to intermediate shell circumferential welds.

2a) The nickel content for STP Unit 2 welds with heat number 90209 (0.11 weight-percent) is per UFSAR Table 5.3-4 and is based on CE NPSD-1039 Rev. 2, "Best Estimate Copper and Nickel in CE Fabricated Reactor Vessel Welds." The use of CE NPSD-1039 was communicated in a letter dated May 19, 1997, "Updated Response to Generic Letter 92-01, Revision 1, Supplement 1," (ST-HL-AE-5628).

2b) The copper values (0.1 weight-percent) used in LRA Table 4.2-2 and Table 4.2-3 for the upper-to-intermediate circumferential weld and the lower shell to lower head circumferential weld are from the reactor vessel specification. The copper values (0.35 weight-percent) used in LRA Table 4.2-4 and Table 4.2-5 for these same welds are conservatively estimated using 10 CFR 50.61(c)(1)(iv)(A). LRA Table 4.2-4 and Table 4.2-5 are revised to use copper value of 0.1 weight-percent consistent with LRA Table 4.2-2 and 4.2-3. Additionally, LRA Section 4.2.3 is revised to state that the end-of-life-extended (EOLE) Cv USE for the welds is greater than 50 ft-lbf.

2c) See 2b above.

Enclosure 2 provides the line-in/line-out revisions to LRA Sections 4.2.2 and 4.2.3 and LRA Tables 4.2-2, 4.2-3, 4.2-4 and 4.2-5.

RAI 4.1-3a

Background:

In the applicant's response to RAI 4.1-3 (November 21, 2011), the applicant indicated that the design basis information in the Updated Final Safety Analysis Report (UFSAR) Section 5.2.3.3.2 provides the applicant's design basis for addressing underclad cracking in the reactor vessel nozzles made from SA-508, Class 2 forging materials. The applicant also stated that the referenced "special evaluation" UFSAR Section 5.3.1.2 does not need to be identified as a TLAA because the regulatory position in NRC RG 1.43 for qualifying clad-to-forging weld qualification tests do not account for an aging mechanism or involve a time-dependent aging parameter. The staff's original request for additional information (RAI) 4.1-3 provides a more detailed background and summary of the staff's initial concern regarding this issue.

Issue:

In its response to RAI 4.1-3, the applicant based its "absence of a TLAA" conclusion for the RV SA-508 Class 2 forging components on the criteria that were established in RG 1.43 and not on the activities or analyses that were implemented in the CLB in order to conform to the weld qualification test criteria in RG 1.43. Nor was the applicant's conclusion based on a comparison of these activities or evaluations to the criteria for identifying TLAAs in 10 CFR 54.3.

Request:

Summarize and describe in sufficient detail the type of tests or evaluations that were performed as part of the CLB in order to meet the recommended weld qualification criteria RG 1.43. If an analysis, evaluation, or calculation was performed as part of the CLB for STP's weld qualification basis, clarify how the applicable document of record compares to each of the six (6) criteria for TLAAs in 10 CFR 54.3, and identify whether the analysis, evaluation or calculation needs to be identified as a TLAA in accordance with 10 CFR 54.21(c)(1). Justify the basis for your determinations and conclusions.

STPNOC Response:

The STPNOC response to this RAI is provided in Reference 5 to this letter. In addition to revising LRA Sections 3.1.2.2.5 and 4.7.4, Tables 4.1-1 and 4.1-2 and adding new Appendix A3.6.5 as discussed in Reference 5 to this letter, Table 3.1.2-1 is revised to disposition intergranular separation in the heat-affected zone (HAZ) of reactor vessel low-alloy steel under austenitic SS cladding as a TLAA in accordance with 10 CFR 54.21(c)(1)(i).

Enclosure 2 provides the line-in revision for changes to LRA Table 3.1.2-1.

SOUTH TEXAS PROJECT, UNITS 1 AND 2
REQUEST FOR ADDITIONAL INFORMATION -
AGING MANAGEMENT, SET 15
(TAC NOS. ME4936 AND ME4937)

RAI B2.1.3-2b

Background:

By letter dated December 15, 2011, the applicant responded to RAI B2.1.3-2a that addresses the inspections for stud insert #30 (also called stud hole insert #30) of Unit 2, the lugs of which were partially rolled. The applicant's response indicates that procedures will be enhanced to perform a remote VT-1 of stud insert #30 concurrent with the volumetric examination once every 10 years. The applicant also indicated that this enhancement will be implemented prior to the period of extended operation (PEO).

By letter dated January 18, 2012, the applicant responded to Request 4 of RAI B2.1.3-2a that addresses the stress analysis requirements of American Society of Mechanical Engineers (ASME) Code Section III for stud insert #30. The applicant's response indicates that the Rotolok Mechanism is designed under all conditions to meet the requirements of the applicable sections of ASME Code, Section III, 1971 edition with addenda through the summer of 1973. However, the applicant did not address the stress analysis results for the emergency conditions. The applicant also stated that due to the nature of the bearing deformation damage, the original stress analysis results were not considered to have changed as the critical cuts and loading did not change. The applicant further indicated that the bearing stress on the non-deformed surfaces of the insert lugs was determined to be limiting consideration.

In response to RAI 4.3-8 dated November 21, 2011, the applicant stated the damage to the stud hole insert is along about 17 percent of the length of the lug and that the damage is radially inward from the location of the maximum usage factor (at the intersection of the lug and the vertical cylinder surface of the insert) such that the bending moment loading on the lugs is not as great at maximum usage factor location as at the damaged location. Therefore, the increase in stress at the maximum usage factor location would be less than 17 percent.

Issue:

The applicant's responses indicate that an additional visual inspection has not been performed on stud insert #30 to confirm no additional reduction in the load bearing surfaces after the damaged stud insert was placed in service in 2007. The applicant's proposed visual inspection schedule for stud insert #30 may delay the successive visual inspections as late as 10 years after entering the PEO; therefore, the absence of additional bearing surface reduction and degradation in the stud insert cannot be confirmed prior to entering the PEO.

The staff also needs baseline information regarding the depth of the damaged areas of the stud insert lugs and clarification as to why the applicant's response dated January 18, 2012, refers to the Unit 1 Stress Report dated October 1977, rather than Unit 2 Stress Reports, in its discussion on the stress analysis for the faulted condition.

The staff further needs to confirm whether the damaged stud insert complies with the stress limits of ASME Code Section III for the emergency conditions. In addition, the applicant's responses do not provide sufficient information to justify why the partially rolled lugs of stud insert #30 do not change the original stress analysis results that were based on the undamaged stud insert lugs.

It is not clear how the applicant determined that the increase in the stress at the maximum usage factor location would be less than 17 percent and how the applicant can determine the increase in stress at the location of maximum usage factor would not result in exceeding the ASME Code design limit of 1.0.

Request:

1. The applicant's proposed schedule for successive visual inspections of stud insert #30, which is once every 10 years during the PEO, appears to delay the successive visual inspections as late as 10 years after entering the PEO. Justify the adequacy of the proposed inspection schedule.

2. Describe the depth of the partially rolled areas of the stud insert lugs as the baseline information for the load bearing surface damage (5.14 in^2, which is 17 percent of the total load bearing surfaces of the stud insert lugs). In addition, describe the characteristics of the transition regions of the partial rolling, which are adjacent to the undamaged surfaces of the lugs, in order to assess the degree of stress concentration due to the damage.

3. Since the damaged stud insert (#30) is in Unit 2, provide the correct Unit 2 references, instead of the Unit 1 references cited in the January 18, 2012, RAI response.

4. Justify why the continued use of the damaged stud insert ensures that the stresses on the component for emergency conditions are bounded by the stress limits of ASME Code Section III such that no additional age-related concerns are present for the period of extended operation.

5. Provide additional information to justify why the partially rolled lugs of stud insert #30 do not invalidate the original stress analysis results and original calculated maximum cumulative usage factor for the stud hole insert for the period of extended operation. As part of the response, provide the maximum acceptable reduction in load bearing surfaces of the stud insert lugs, which complies with the stress limits in ASME Code

   ASME Code Section III, assuming the same type of lug damage observed in April 2007. In addition, compare the observed 17 percent reduction in the load bearing surfaces with the maximum acceptable reduction in the load bearing surfaces of the lugs.

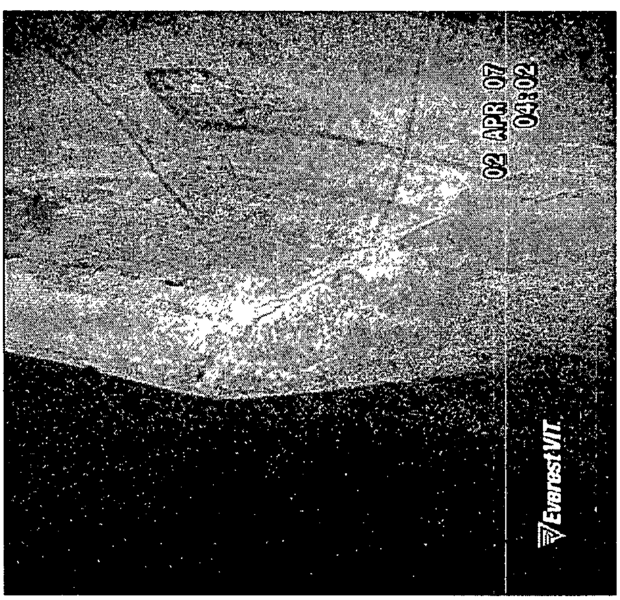
6. Justify how it was determined that the increase in the stress at the maximum usage factor location would be less than 17 percent. Provided that this justification is acceptable, justify that the increase in stress at the location of maximum usage factor would not exceed the Code design limit of 1.0 through the period of extended operation.

STPNOC Response:

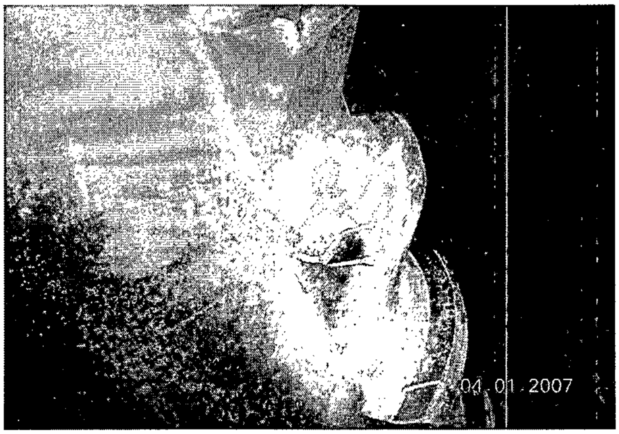
1. Remote VT-1 of stud insert at stud location #30 is performed once every 10 years starting with the current (Third) 10-year ASME Section XI inspection interval. STP has just entered the Third Interval. These inspections will continue through the period of extended operation. LRA Commitment 38 is revised and new Commitment 42 is added to clarify the implementation schedule for performing the inspections.

2. The measured depth of the rolled indentation of the stud insert is approximately 0.005"-0.010" deep and 0.5"-0.75" wide (circumferentially). All insert lugs have similar damage. The rolled area transition is smooth to touch and is similar to the transition on the stud transition area (see attached photo). A visual inspection was performed on the stud insert and the inspector rubbed a rubber glove over the transition. The rubber glove was not damaged and did not snag on the insert damage. Therefore, it is concluded that the transition is smooth and does not have upset metal in the area.

   Insert Lug at stud location #30 (all lugs on the insert have similar indentation)



Bottom Lugs of stud location #30



3. In reference to the January 18, 2012 RAI response letter (Ref: ML12020A072, NOC-AE-12002779), the following Unit 2 references are provided:

   - When referring to the Addendum to the Combustion Engineering (CE) Stress Report, dated October 1986, in the January 18, 2012 letter regarding documentation of the comprehensive thermal/stress analysis for normal and upset conditions using a 3-D finite element model, the Unit 2 reference is:

     Westinghouse Report MED-PCE-6279, "Addendum to the Combustion Engineering Final Stress Report for the South Texas Unit No. 2 Reactor Vessel," Section 11, June 24, 1988

     The maximum stress intensity range remains at 98.85 ksi with the ASME Code Section III allowable at 120 ksi.

   - When referring to CE Stress Report for South Texas Project Unit 1, dated October 1977, in the January 18, 2012 letter regarding the maximum faulted condition stress for Rotolok stud system for the primary stress resulting from the maximum faulted condition transient (control rod ejection), the Unit 2 reference is:

     Combustion Engineering Report, CENC-1354, "Analytical Report for South Texas Project No. 2 Houston Lighting and Power Company," January 1979

     The maximum faulted condition transient (control rod ejection) reported as 78.70 ksi on Page A-192 of the report is unchanged.

4. According to ASME Section III, Paragraph NB-3224, the rules used for evaluating other conditions shall be used for evaluating Emergency Conditions except as modified as follows in the subsequent paragraphs:

   NB-3224.1 Primary Stress Limits. The primary stress of NB-3221 for Design Conditions shall be satisfied using a design stress intensity value (S_m) equal to the greater of 120 percent of the tabulated S_m value or 100 percent the tabulated yield strength, with both values taken at the appropriate temperature.

   NB-3224.2 External Pressure. This is not applicable for the stud hole inserts.

   NB-3224.3 Special Stress Limits. The permissible values shall be taken as 120 percent of the Design Condition limits.

   NB-3224.4 Secondary and Peak Stresses. The requirement for Normal and Upset Conditions need not be satisfied for Emergency Conditions.

   NB-3224.5 Fatigue Requirements. Emergency Conditions need not be considered in the fatigue analysis.

   NB-3226.6 Deformation Limits. Any deformation limits prescribed in the design specifications shall be considered. No deformation limits are prescribed for the stud hole inserts in the STP reactor vessel design specifications.

   The stud hole inserts are threaded sleeves in the reactor vessel flange stud holes and therefore, are not subject to the Stress Limits for Bolts in NB-3230.

   The limiting stud hole insert bearing stress occurs during the Normal Condition Plant Heatup transient for which the stud tensile load exceeds any Emergency Condition stud tensile load. Review of the Nuclear Steam Supply System (NSSS) design transients applicable to STP Unit 2 reveals that the Emergency Condition design transients are identified as Small Loss of Coolant Accident, Small Steam Break, and Complete Loss of Flow. The transient pressure variation figures for these design transients show that the maximum Emergency Condition reactor coolant system pressure is 2417 psig during the last part of the Small Steam Break transient. The maximum internal pressure resulting from all of the Emergency Condition transients is less than the Design Pressure of 2485 psig. Therefore, the primary stress intensities in the stud hole inserts resulting from the Emergency Conditions are less than the primary stress intensities for the Design Condition. The stress limits for Emergency Conditions are greater than the stress limits for the Design Condition. The primary stress intensities for Emergency Conditions are bounded by the Design Condition and the applicable primary stress limits for Emergency Conditions.

5. According to Report CENC-1354, Page A-866, the maximum bolt tensile load applied to the lugs of a single stud hole insert is 2,593,000 lb following a plant heatup at 100°F/hr. The bearing area for each undamaged lug is 1.44 in^2. With the 17 % reduction in bearing area due to the damage to the lugs in stud location #30, the remaining bearing area for each lug is 1.195 in^2. The maximum bearing stress on each lug is then:

2,593 kips/(7 x 3 x 1.195 in^2)] = 103.33 ksi

- where 7 is the number of rows of lugs on each insert, 3 is the number of lugs per row, and 1.195 in^2 is the remaining bearing area for each damaged lug.

This result is compared to the tabulated yield strength for the SA-540, Grade B24, Class 3 stud hole insert material at a temperature of 350°F, which is the temperature of the stud hole inserts at the end of plant heatup according to Westinghouse Report MED-PCE-6279. This tabulated yield strength at 350°F is 118.5 ksi according to Table I-13.3 in the ASME Section III, Appendices, 1986 Edition.

For the bearing stress to reach the 118.5 ksi limit, the bearing area would have to be reduced to:

A = 2,593 kips/(7 x 3 x 118.5 ksi) = 1.042 in^2.

This would amount to a further 10.6 % reduction from the original 1.44 in^2 lug bearing area and a 13% reduction from the current 1.195 in^2 area.

See the response to request for additional information #6 below for fatigue justification.

6. Both the lugs of the stud and the lugs of the insert are designed with a 0.25" x 45° chamfer that positions the edge of the bearing area away from the wall of the insert to protect the fillet radii of the insert lugs from bearing deformation. Therefore, the bearing deformation is positioned off the fillet radius and at least 0.25" from the lug intersection with the insert inside diameter (ID) wall where the maximum usage factor is calculated. Contact between the ends of the stud lugs with the insert ID would not permit the edges of bearing deformation to be closer.

   Since the edge of the bearing deformation is positioned inside the fillet radii of the insert lugs, the bearing damage does not create higher peak stress intensities that would cause the cumulative fatigue usage factor (CUF) to increase as result of additional stress concentration. The edge of the bearing deformation that could contribute a stress concentration factor that would magnify the peak stress intensities on the lugs is only along approximately 17% of the width of the lugs parallel to the lug intersection with the insert ID wall. Furthermore, with the inserts properly engaged with the stud inserts, the deformed sections of the lugs will see reduced stress due to depression of the surface in these sections. Also, the bending stress is less at the edge of the bearing deformation since it is radially inward from the location of the maximum usage factor at the lug/ID wall juncture, and the moment arm is reduced.

   The reported maximum fatigue usage factor for the stud hole insert of 0.8852 in Report No. MED-PCE-6279 is conservative considering the way in which the maximum CUF at lug no. 7 is calculated. The largest contribution to the 0.8852 CUF is due to the pairing of Cold Hydrostatic Test and Unit Loading at 5 percent of Full Power per minute. The calculation uses 13,177 as the number of cycles when only 10 occurrences of Cold Hydrostatic Test need be considered. Using 10 cycles as defined by Cold Hydrostatic Test reduces the calculated CUF by 0.470. This correction reduces the CUF to less than 0.4200. Further procedural corrections permitted by ASME Section III, NB-3222.4 and elimination of almost any effect of Unit Loading and Unloading at 5 percent per minute due to base loading and Upper Head Temperature Reduction reduces the calculated CUF to less than 0.4000. This reduction in the maximum CUF leaves margin under the ASME Code limit of 1.0 to apply a higher composite stress concentration factor than the 1.85 that should be considered for the lug fillet radii, if needed. However, such a modification to account for the edge of the bearing deformation is judged to be not warranted because the results in Report No. MED-PCE-6279 are already conservative. According to Section 11.14.3 of the report, the fillet radii could not be modeled in the 3-D finite element analysis, and the results already include high stress concentration so that the primary plus secondary plus peak stresses can be used directly. No further fatigue strength reduction factor is needed for the fatigue evaluation of the stud hole insert lugs, even with the bearing deformation.

The reported 0.8852 cumulative usage factor for stud hole insert lug no. 7 in STP Unit 2 remains conservative as the maximum CUF for the damaged stud hole insert by a factor of 2.

Enclosure 3 provides the line-out revision to Commitment 38 and line-in of new Commitment 42.

Enclosure 2

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

List of Revised LRA Sections

RAI	Affected LRA Section
4.2.2-1	Section 4.2.2 Section 4.2.3 Table 4.2-2 Table 4.2-3 Table 4.2-4 Table 4.2-5
4.1-3a	Table 3.1.2-1

4.2.2 Pressurized Thermal Shock

Summary Description

10 CFR 50.61(b)(1) provides rules for protection against pressurized thermal shock (PTS) events for pressurized water reactors. Licensees are required to perform an updated assessment of the projected values of PTS reference temperature (RT_{PTS}) whenever there is a significant change in projected values of RT_{PTS}, or upon a request for a change in the expiration date for operation of the facility.

The license renewal rule 10 CFR 54.4(a)(3) also requires that the licensee evaluate those structures, systems, and components (SSCs) relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with the Commission's regulations for PTS.

10 CFR 50.61(c) provides two methods for determining RT_{PTS}. These methods are also described as Positions 1 and 2 in Regulatory Guide 1.99, *Radiation Embrittlement of Reactor Vessel Materials.* Position 1 applies for material that *does not* have credible surveillance data available and Position 2 is used for material that *does* have two or more credible surveillance data sets available. The adjusted reference temperatures are calculated for both Positions 1 and 2 by following the guidance in Regulatory Guide 1.99 (Sections 1.1 and 2.1, respectively), using the copper and nickel content of STP beltline materials, and the EOLE fluence projections.

10 CFR 50.61(b)(2) establishes screening criteria for RT_{PTS} as 270°F for plates, forgings, and axial welds and 300°F for circumferential welds. If the RT_{PTS} does not exceed the PTS screening criteria, then only the reactor pressure vessel is relied on to demonstrate compliance with the 10 CFR 50.61, the PTS rule.

Analysis

The original response to the issuance of the PTS rule, 10 CFR 50.61, by STP indicated that the projected RT_{PTS} for both units do not exceed the PTS screening criteria (270°F for plates, forgings, and axial welds; and 300°F for circumferential welds), based on a 40 year, 32 EFPY life.

The most recent coupon examination results for both units show that the shift in RT_{NDT} in plate and weld materials are in good agreement with or less than the Regulatory Guide 1.99 Revision 2 predictions for Units 1 and 2. The results demonstrate that the Regulatory Guide 1.99 predictions provide a conservative means to satisfy the requirement of 10 CFR 50.61; thus providing assurance of the reactor vessel integrity.

Unit 1

The data from the most recently withdrawn surveillance capsule, Capsule V, were deemed credible. RT_{PTS} values were generated for beltline and extended beltline region materials of the Unit 1 reactor vessel for fluence values at EOLE (54 EFPY). The Unit 1 RT_{NDT} results from Capsule V indicated measured mean 30 ft-lb transition temperatures of -12.17°F, 17.78°F, and -29.41°F for longitudinal plate coupons, transverse plate coupons, and the weld metal respectively. The Capsule V material is from intermediate shell R1606-2.

The RT_{PTS} values for the Unit 1 beltline materials are provided in Table 4.2-2. The limiting material for Unit 1 is the intermediate shell R1606-3 with a projected EOLE RT_{PTS} value of 83.6°F. The projected RT_{PTS} values for EOLE meet the 10 CFR 50.61 screening criteria.

The extended beltline materials that are expected to receive fluence values greater than 1 x 10^{17} n/cm^2 (E>1.0 MeV) were also evaluated. The RT_{PTS} values were shown to meet the 10 CFR 50.61 screening criteria. <u>The fluence projections for the nozzle (upper) shell to intermediate shell circular weld and lower shell to lower head torus circular weld bound the other materials above and below the beltline.</u>

Unit 2

The data from the most recently withdrawn surveillance capsule, Capsule U, was deemed credible. RT_{PTS} values were generated for beltline and extended beltline region materials of the Unit 2 reactor vessel for fluence values at EOLE (54 EFPY). The Unit 2 RT_{NDT} results from Capsule U indicated measured mean 30 ft-lb transition temperatures of -10.49°F, 22.23°F, and 5.88°F for longitudinal plate coupons, transverse plate coupons, and the weld metal respectively. The Capsule U material is from intermediate shell R2507-1.

The RT_{PTS} values for the Unit 2 beltline materials are provided in Table 4.2-3. The limiting material for Unit 2 is the intermediate shell R2507-2 with a projected EOLE RT_{PTS} value of 63.7°F. The projected RT_{PTS} values for EOLE meet the 10 CFR 50.61 screening criteria.

The extended beltline materials that are expected to receive fluence values greater than 1 x 10^{17} n/cm^2 (E>1.0 MeV) were also evaluated. The RT_{PTS} values were shown to meet the 10 CFR 50.61 screening criteria. <u>The fluence projections for the nozzle (upper) shell to intermediate shell circular weld and lower shell to lower head torus circular weld bound the other materials above and below the beltline.</u>

Disposition: Projection, 10 CFR 54.21(c)(1)(ii)

The RT_{PTS} values were revised with projections to the end of the period of extended operation. Therefore, these TLAAs are dispositioned in accordance with 10 CFR 54.21(c)(1)(ii).

4.2.3 Upper-Shelf Energy (USE)

Summary Description

Per Regulatory Guide 1.99, *Radiation Embrittlement of Reactor Vessel Materials*, the Charpy upper-shelf energy (C_V USE) is assumed to decrease as a function of fluence and copper content. Figure 2 of the guide determines this magnitude of decrease when surveillance data is not used (Position 1.2). In addition, if surveillance data is to be used (Position 2.2), the decrease in upper shelf energy may be obtained by plotting the reduced plant surveillance data on Figure 2 of the guide and fitting the data with a line drawn parallel to the existing lines as the upper bound of all the data. This line can then be used in preference to the existing line. The C_V USE can be predicted using the corresponding ¼T fluence projection, the copper content of the beltline materials, and the results of the capsules tested to date using Figure 2 of the guide. The fluence at the ¼T depth (f) is determined using the clad / base metal fluence (fsurf) and the depth of the desired location in inches.

10 CFR 50, Appendix G, Section IV.A.1.a requires that the reactor vessel beltline materials must have a C_V USE of no less than 75 ft-lb initially, and must maintain C_V USE throughout the life of the vessel of no less than 50 ft-lb unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of C_V USE will provide margins of safety against fracture equivalent to those required by ASME Section XI, *Rules for In-Service Inspection of Nuclear Power Plant Components, Appendix G.*

Analysis

The most recent coupon examination results for both units show that the decline in C_V USE in plate and weld materials are less than originally predicted by Regulatory Guide 1.99 Revision 2 for Units 1 and 2. The results demonstrate that the Regulatory Guide 1.99 predictions provide a conservative means to satisfy the requirements of 10 CFR 50, Appendix G; thus providing assurance of the reactor vessel integrity.

Unit 1

The C_V USE results from Unit 1 surveillance Capsule V indicated a mean Charpy V-notch C_V USE of 131 ft-lbf, 106 ft-lbf and 86 ft-lbf for longitudinal plate coupons, transverse plate coupons, and the weld metal respectively. The data were determined to be credible, however the data were not included in the EOLE C_V USE projections.

To support operation during the period of extended operation, these values were projected to 54 EFPY of operation. The EOLE C_V USE values for the Unit 1 beltline materials are provided in Table 4.2-4. The limiting value was 71 ft-lbf for intermediate shell R1606-2.

The extended beltline nozzle and shell materials that are expected to receive fluence values greater than 1 x 10^{17} n/cm^2 (E>1.0 MeV) were also evaluated and confirm an EOLE C_V USE that is greater than 50 ft-lbf. <u>The fluence projections for the nozzle</u>

(upper) shell to intermediate shell circular weld and lower shell to lower head torus circular weld bound the other materials above and below the beltline.~~The weld material is addressed below.~~

Unit 2

The C_V USE results from Unit 2 surveillance Capsule U indicated a mean Charpy V-notch C_V USE of 138 ft-lbf, 98 ft-lbf, and 97 ft-lbf for longitudinal plate coupons, transverse plate coupons, and the weld metal respectively. The Surveillance Capsule U results for Unit 2 were deemed credible, however the data were not included in the EOLE C_V USE projections.

To support operation during the period of extended operation, these values were projected to 54 EFPY of operation. The EOLE C_V USE values for the Unit 2 beltline materials are provided in Table 4.2-5. The limiting value was 75 ft-lbf for lower shell longitudinal weld E3.12.

The extended beltline nozzle and shell materials that are expected to receive fluence values greater than 1 x 10^{17} n/cm^2 (E>1.0 MeV) were also evaluated. The bottom head torus, R3020-1, was identified to have a projected EOLE C_V USE value of 76 ft-lbf. The RT_{PTS} is more limiting than the beltline material. However the C_V USE still meets the 10 CFR 50, Appendix G 50 ft-lbf criterion. The fluence projections for the nozzle (upper) shell to intermediate shell circular weld and lower shell to lower head torus circular weld bound the other materials above and below the beltline.~~The weld material is addressed below~~

~~*Units 1 and 2 Extended Beltline Welds*~~

~~The estimated limiting EOL C_V USE of Unit 1 and 2 extended beltline welds is 49 ft-lbf. However the assumptions used to evaluate these welds include several conservativisms: (1) The assumed unirradiated C_V USE is the lowest acceptable value; (2) The assumed Cu value is the maximum acceptable value; and (3) the percent decrease in C_V USE is based on 10^{18} n/cm^2 which is the lowest value in Figure 2 of RG 1.99, but the actual fluence is less than 4 x 10^{17} n/cm^2. These conservativisms can account for the 1 ft-lbf (2%) difference between the criterion and the projected value. Extrapolation of the upper limit line in Figure 2 of RG 1.99 to 4 x 10^{17} n/cm^2 increases the predicted EOL C_V USE by greater than 2%. Therefore it was determined that the C_V USE of these welds will remain above the 50 ft-lbf criterion.~~

Disposition: Projection, 10 CFR 54.21(c)(1)(ii)

The C_V USE values were re-evaluated with projections to the end of the period of extended operation. Therefore, these TLAAs are dispositioned in accordance with 10 CFR 54.21(c)(1)(ii). The re-evaluations demonstrated that the C_V USE in the limiting material of each unit will remain above the 10 CFR 50 Appendix G acceptance criteria of 50 ft-lbf.

Table 4.2-2 STP Unit 1 Vessel Limiting 54 EFPY RT_{PTS}

Material Description			Chemistry			Fluence		Margin			54 EFPY RT_{NDT}				Ext. Belt-line (Y/N)
Component	**Heat Number.**	**Base Metal / Flux Type**	**Cu (%)**	**Ni (%)**	**CF**	**Fluence (10^{19} n/cm^2)**	**FF**	σ_I	σ_Δ	**Margin**	**Initial RT_{NDT}**	ΔRT_{NDT}	**EOL RT_{NDT} (w/out margin)**	RT_{PTS}	
Base Metals															
Inlet Nozzle R1613-1	-	SA508 CL. 2	0.35	0.8	241	0.0345	0.24	17	17	48.1	-10	57	47.4	95.5	Y
Inlet Nozzle R1613-2	-	SA508 CL. 2	0.35	0.82	244.1	0.0345	0.24	17	17	48.1	0	58	58.1	106.2	Y
Inlet Nozzle R1613-3	-	SA508 CL. 2	0.09	0.79	58	0.0345	0.24	0	6.9	13.8	-20	14	-6.2	7.6	Y
Inlet Nozzle R1613-4	-	SA508 CL. 2	0.35	0.85	248.8	0.0345	0.24	17	17	48.1	20	59	79.2	127.3	Y
Outlet Nozzle R1614-1	-	SA508 CL. 2	0.35	0.66	220.7	0.0345	0.24	17	17	48.1	10	53	62.6	110.6	Y
Outlet Nozzle R1614-2	-	SA508 CL. 2	0.35	0.71	228.0	0.0345	0.24	17	17	48.1	0	54	54.3	102.4	Y
Outlet Nozzle R1614-3	-	SA508 CL. 2	0.35	0.69	225.1	0.0345	0.24	17	17	48.1	-30	54	23.6	71.7	Y
Outlet Nozzle R1614-4	-	SA508 CL. 2	0.35	0.9	256.5	0.0345	0.24	17	17	48.1	10	61	71.1	119.2	Y
Nozzle Shell R1607-1	-	SA533B CL. 1	0.08	0.62	51	0.0345	0.24	0	6.1	12.1	50	12	62.1	74.3	Y

Table 4.2-2 STP Unit 1 Vessel Limiting 54 EFPY RT_{PTS}

Material Description			**Chemistry**			**Fluence**		**Margin**			**54 EFPY RT_{NDT}**				**Ext. Belt-line (Y/N)**
Component	**Heat Number.**	**Base Metal / Flux Type**	**Cu (%)**	**Ni (%)**	**CF**	**Fluence (10^{19} n/cm^2)**	**FF**	**σ_I**	**σ_Δ**	**Margin**	**Initial RT_{NDT}**	**ΔRT_{NDT}**	**EOL RT_{NDT} (w/out margin)**	**RT_{PTS}**	
Nozzle Shell R1607-2	-	SA533B CL. 1	0.08	0.66	51	0.0345	0.24	0	6.1	12.1	50	12	62.1	74.3	Y
Nozzle Shell R1607-3	-	SA533B CL. 1	0.07	0.6	44	0.0345	0.24	0	5.2	10.5	30	10	40.5	51.0	Y
Inter. Shell R1606-1	B-8120-2	SA533B CL. 1	0.04	0.63	26	2.85	1.28	0	16.6	33.2	10	33	43.2	76.5	N
Inter. Shell R1606-2	B-8120-1	SA533B CL. 1	0.04	0.61	26	2.85	1.28	0	16.6	33.2	0	33	33.2	66.5	N
	S/C		-	-	26.6	2.85	1.28	0	8.5	17.0	0	34	34.0	51.0	
Inter. Shell R1606-3	C-4326-2	SA533B CL. 1	0.05	0.62	31	2.85	1.28	0	17.0	34.0	10	40	49.6	83.6	N
Lower Shell R1622-1	B-9566-2	SA533B CL. 1	0.05	0.61	31	3.86	1.35	0	17.0	34.0	-30	42	11.8	45.8	N
Lower Shell R1622-2	B-9575-2	SA533B CL. 1	0.07	0.64	44	3.86	1.35	0	17.0	34.0	-30	59	29.3	63.3	N
Lower Shell R1622-3	B-9575-1	SA533B CL. 1	0.05	0.66	31	3.86	1.35	0	17.0	34.0	-30	42	11.8	45.8	N
Bottom Head Torus R1617-1	-	SA533B CL. 1	0.14	0.67	101.8	0.0341	0.24	0	12.0	24.1	-30	24	-5.9	18.1	Y
Bottom Head Dome R1618-1	-	SA533B CL. 1	0.08	0.67	51	0.0341	0.24	0	6.0	12.1	-30	12	-17.9	-5.9	Y

Table 4.2-2 STP Unit 1 Vessel Limiting 54 EFPY RT_{PTS}

Material Description			Chemistry			Fluence		Margin			54 EFPY RT_{NDT}				Ext. Belt-line (Y/N)
Component	Heat Number.	Base Metal / Flux Type	Cu (%)	Ni (%)	CF	Fluence (10^{19} n/cm^2)	FF	σ_I	σ_Δ	Margin	Initial RT_{NDT}	ΔRT_{NDT}	EOL RT_{NDT} (w/out margin)	RT_{PTS}	
Welds															
Inlet / Outlet nozzle to shell welds(1)	-	Linde 0091	0.1	1	135	<0.0345	<0.24	17	16.1	46.8	-56	< 32	< -23.9	< 22.9	Y
Nozzle (Upper) shell long. welds	-	Linde 0091	0.1	1	135	≤0.0345	≤0.24	17	16.1	46.8	-56	≤ 32	≤ -23.9	≤ 22.9	Y
Nozzle (Upper) shell to Inter. shell circ. ~~Weld~~weld	-	Linde 0091	0.1	1	135	0.0345	0.24	17	16.1	46.8	-56	32	-23.9	22.9	Y
Inter. Shell long. weld G1.70	89476	Linde 0091	0.02	0.07	27	1.53	1.12	0	15.1	30.2	-50	30	-19.8	10.4	N
	S/C Data	-	-	-	30.4	1.53	1.12	0	8.5	17.0	-50	34	-16.0	1.0	
Lower shell long. Weld G1.70	89476	Linde 0091	0.02	0.07	27	3.86	1.35	0	18.2	36.4	-50	36	-13.6	22.8	N
	S/C	-	-	-	30.4	3.86	1.35	0	10.2	20.5	-50	41	-9.0	11.5	

Table 4.2-2 STP Unit 1 Vessel Limiting 54 EFPY RT_{PTS}

Material Description			Chemistry			Fluence		Margin			54 EFPY RT_{NDT}				Ext. Belt-line (Y/N)
Component	Heat Number.	Base Metal / Flux Type	Cu (%)	Ni (%)	CF	Fluence (10^{19} n/cm^2)	FF	σ_I	σ_Δ	Margin	Initial RT_{NDT}	ΔRT_{NDT}	EOL RT_{NDT} (w/out margin)	RT_{PTS}	
Inter. to lower circ. weld E3.13	89476	Linde 124	0.02	0.07	27	2.84	1.28	0	17.2	34.5	-70	34	-35.5	-1.0	N
	S/C	-	-	-	30.4	2.84	1.28	0	9.7	19.4	-70	39	-31.2	-11.7	
Lower shell to lower head torus circ. ~~Weld~~weld	-	Linde 124	0.1	1	135	0.0341	0.24	17	16.0	46.6	-56	32	-24.1	22.6	Y
Lower head torus long. weld	AAOHF	-	0.02	0.96	27	≤0.0341	≤0.24	0	3.25	6.5	-80	≤ 6.5	≤ -73.5	≤ -67	Y
Lower head torus to dome circ. weld	-	Linde 0091	0.1	1	135	<0.0341	<0.24	17	16.0	46.6	-56	< 32	< -24.1	<22.6	Y

[1] Per the Weld Inspection Forms, the Unit 1 inlet nozzle to shell circumferential weld 105-121A was fabricated using manual E-8018 type welds. The initial properties (RT_{NDT} and USE) for the E-8018 type welds are bounded by the generic Linde 0091 flux type weld properties. Therefore, the generic Linde 0091 flux type initial RT_{NDT} and USE values are used for this weld.

Table 4.2-3 STP Unit 2 Vessel Limiting RT_{PTS}

Material Description			Chemistry			Fluence		Margin			54 EFPY RT_{NDT}				Ext. Belt-line (Y/N)
Component	Heat Number.	Base Metal / Flux Type	Cu (%)	Ni (%)	CF	Fluence (10^{19} n/cm^2)	FF	σ_I	σ_Δ	Margin	Initial RT_{NDT}	ΔRT_{NDT}	EOL RT_{NDT} (w/out margin)	RT_{PTS}	
Base Metals															
Inlet Nozzle R2011-1	-	SA508 CL. 2	0.1	0.81	67	0.0328	0.23	0	7.7	15.5	-40	15	-24.5	-9.0	Y
Inlet Nozzle R2011-2	-	SA508 CL. 2	0.1	0.84	67	0.0328	0.23	0	7.7	15.5	-20	15	-4.5	11.0	Y
Inlet Nozzle R2011-3	-	SA508 CL. 2	0.12	0.84	86	0.0328	0.23	0	9.9	19.9	-20	20	-0.1	19.8	Y
Inlet Nozzle R2011-4	-	SA508 CL. 2	0.11	0.83	77	0.0328	0.23	0	8.9	17.8	-20	18	-2.2	15.6	Y
Outlet Nozzle R2012-1	-	SA508 CL. 2	0.35	0.72	229.4	0.0328	0.23	17	17.0	48.1	10	53	63.1	111.1	Y
Outlet Nozzle R2012-2	-	SA508 CL. 2	0.35	0.68	223.6	0.0328	0.23	17	17.0	48.1	10	52	61.7	109.8	Y
Outlet Nozzle R2012-3	-	SA508 CL. 2	0.35	0.67	222.2	0.0328	0.23	17	17.0	48.1	0	51	51.4	99.5	Y
Outlet Nozzle R2012-4	-	SA508 CL. 2	0.35	0.68	223.6	0.0328	0.23	17	17.0	48.1	10	52	61.7	109.8	Y
Nozzle Shell R2505-1	-	SA533B CL. 1	0.05	0.66	31	0.0328	0.23	0	3.6	7.2	0	7	7.2	14.3	Y
Nozzle Shell R2505-2	-	SA533B CL. 1	0.07	0.64	44	0.0328	0.23	0	5.1	10.2	0	10	10.2	20.4	Y
Nozzle Shell R2505-3	-	SA533B CL. 1	0.05	0.65	31	0.0328	0.23	0	3.6	7.2	-10	7	-2.8	4.3	Y

Table 4.2-3 STP Unit 2 Vessel Limiting RT_{PTS}

Material Description			Chemistry			Fluence		Margin			54 EFPY RT_{NDT}				Ext. Belt-line (Y/N)
Component	**Heat Number.**	**Base Metal / Flux Type**	**Cu (%)**	**Ni (%)**	**CF**	**Fluence (10^{19} n/cm^2)**	**FF**	**σ_I**	**σ_Δ**	**Margin**	**Initial RT_{NDT}**	**ΔRT_{NDT}**	**EOL RT_{NDT} (w/out margin)**	**RT_{PTS}**	
Inter. Shell R2507-1	NR 62 067-1	SA533B CL. 1	0.04	0.65	26	2.86	1.28	0	16.6	33.3	-10	33	23.3	56.5	N
	Using S/C Data	SA533B CL. 1	-	-	28.9	2.86	1.28	0	8.5	17.0	-10	37	27.0	44.0	
Inter. Shell R2507-2	NR 62 230-1	SA533B CL. 1	0.05	0.64	31	2.86	1.28	0	17	34.0	-10	40	29.7	63.7	N
Inter. Shell R2507-3	NR 62 248-1	SA533B CL. 1	0.05	0.61	31	2.86	1.28	0	17	34.0	-40	40	-0.3	33.7	N
Lower Shell R3022-1	NR 64 647-1	SA533B CL. 1	0.03	0.63	20	3.72	1.34	0	13.4	26.8	-30	27	-3.2	23.6	N
Lower Shell R3022-2	NR 64 627-1	SA533B CL. 1	0.04	0.61	26	3.72	1.34	0	17	34.0	-40	35	-5.2	28.8	N
Lower Shell R3022-3	NR 64 445-1	SA533B CL. 1	0.04	0.6	26	3.72	1.34	0	17	34.0	-40	35	-5.2	28.8	N
Bottom Head Torus R3020-1	-	SA533B CL. 1	0.11	0.65	74.5	0.0343	0.24	0	8.9	17.7	40	18	57.7	75.4	Y
Bottom Head Dome R3021-1	-	SA533B CL. 1	0.09	0.64	58	0.0343	0.24	0	6.9	13.8	-60	14	-46.2	-32.5	Y

Table 4.2-3 STP Unit 2 Vessel Limiting RT_{PTS}

Material Description			Chemistry			Fluence		Margin			54 EFPY RT_{NDT}				Ext. Belt-line (Y/N)
Component	**Heat Number.**	**Base Metal / Flux Type**	**Cu (%)**	**Ni (%)**	**CF**	**Fluence (10^{19} n/cm^2)**	**FF**	σ_I	σ_Δ	**Margin**	**Initial RT_{NDT}**	**ΔRT_{NDT}**	**EOL RT_{NDT} (w/out margin)**	**RT_{PTS}**	
Welds															
Inlet / Outlet nozzle to shell welds	-	Linde 124	0.1	1	135	<0.0328	<0.23	17	16.1	46.8	-56	< 31	< -24.8	<21.4	Y
Nozzle (Upper) shell long. welds	-	Linde 0091	0.1	1	135	≤0.0328	≤0.23	17	16.1	46.8	-56	≤ 31	≤ -24.8	≤21.4	Y
Nozzle (Upper) shell to Inter. shell circ. weld ~~seam~~	-	Linde 124	0.1	1	135	0.0328	0.23	17	15.6	46.2	-56	31	-24.8	21.4	Y
Inter. shell long. weld seams G3.02	90209	Linde 0091	0.04	0.11	54	1.68	1.14	0	28	56.0	-70	62	-8.3	47.7	N
	Using S/C Data	-	-	-	8.4	1.68	1.14	0	2.4	4.8	-70	10	-60.4	-55.6	
Lower shell long. weld seams E3.12	90209	Linde 124	0.04	0.11	54	3.72	1.34	0	28	56.0	-70	72	2.4	58.4	N
	Using S/C Data	-	-	-	8.4	3.72	1.34	0	2.8	5.6	-70	11	-58.7	-53.1	

Table 4.2-3 STP Unit 2 Vessel Limiting RT_{PTS}

Material Description			Chemistry			Fluence		Margin			54 EFPY RT_{NDT}				Ext. Belt-line (Y/N)
Component	**Heat Number.**	**Base Metal / Flux Type**	**Cu (%)**	**Ni (%)**	**CF**	**Fluence (10^{19} n/cm^2)**	**FF**	σ_I	σ_Δ	**Margin**	**Initial RT_{NDT}**	ΔRT_{NDT}	**EOL RT_{NDT} (w/out margin)**	RT_{PTS}	
Inter. to lower circ. weld seam E3.12	90209	Linde 124	0.04	0.11	54	2.84	1.28	0	28	56.0	-70	69	-1.0	55.0	N
	Using S/C Data	-	-	-	8.4	2.84	1.28	0	2.7	5.4	-70	11	-59.3	-53.9	
Lower shell to lower head torus circ. ~~Weld~~weld	-	Linde 124	0.1	1	135	0.0343	0.24	17	16.0	46.7	-56	32	-24.0	22.8	Y
Lower head torus long. weld	LAOBF	-	0.025	0.01	22	≤0.0343	≤0.24	0	2.5	5	-70	≤ 5	≤ -65	≤ -60	Y
Lower head torus to dome circ. weld	-	Linde 124	0.1	1	135	< 0.0343	< 0.24	17	16.0	46.7	-56	< 32	< -24.0	< 22.8	Y

Table 4.2-4 STP Unit 1 Reactor Vessel Material C_V USE

Material	Chemistry		54 EFPY C_V USE at the Clad / Base Metal Interface					Ext.
Component	Cu (%)	Fluence (10^{19} n/cm^2)	NPWD[(1)] (ft - lb)	% Decrease in C_V USE	EOL NPWD (ft - lb)	PWD[(2)] (ft - lb)	EOL PWD (ft - lb)	Beltline (Y/N)
Base Metals								
Inlet Nozzle R1613-1	0.35	0.0345	140	26%	104	-	-	Y
Inlet Nozzle R1613-2	0.35	0.0345	130.5	26%	97	-	-	Y
Inlet Nozzle R1613-3	0.09	0.0345	175	11%	156	-	-	Y
Inlet Nozzle R1613-4	0.35	0.0345	128	26%	95	-	-	Y
Outlet Nozzle R1614-1	0.35	0.0345	106	26%	78	-	-	Y
Outlet Nozzle R1614-2	0.35	0.0345	114	26%	84	-	-	Y
Outlet Nozzle R1614-3	0.35	0.0345	129	26%	95	-	-	Y
Outlet Nozzle R1614-4	0.35	0.0345	118	26%	87	-	-	Y
Nozzle Shell R1607-1	0.08	0.0345	89	11%	79	-	-	Y
Nozzle Shell R1607-2	0.08	0.0345	85	11%	76	-	-	Y
Nozzle Shell R1607-3	0.07	0.0345	82	11%	73	-	-	Y
Inter. Shell R1606-1	0.04	2.85	109.5	25%	82	130	95	N
Inter. Shell R1606-2	0.04	2.85	94	25%	71	119	87	N
Inter. Shell R1606-3	0.05	2.85	105.5	25%	79	132	96	N
Lower Shell R1622-1	0.05	3.86	111	27%	81	143	104	N
Lower Shell R1622-2	0.07	3.86	122	27%	89	149	109	N
Lower Shell R1622-3	0.05	3.86	127	27%	93	148	108	N
Bottom Head Torus R1617-1	0.14	0.0341	143	14%	123	-	-	Y
Bottom Head Dome R1618-1	0.08	0.0341	128	11%	114	-	-	Y

Table 4.2-4 STP Unit 1 Reactor Vessel Material C_V USE

Material	Chemistry		54 EFPY C_V USE at the Clad / Base Metal Interface					Ext. Beltline (Y/N)
Component	**Cu (%)**	**Fluence (10^{19} n/cm^2)**	**NPWD[1] (ft - lb)**	**% Decrease in C_V USE**	**EOL NPWD (ft - lb)**	**PWD[2] (ft - lb)**	**EOL PWD (ft - lb)**	
Welds								
Inlet / Outlet nozzle to shell welds[3]	0.10	< 0.0345	101	< 14%	> 87	-	-	Y
Nozzle (Upper) shell long. welds	0.10	≤ 0.0345	101	≤ 14%	≥ 87	-	-	Y
Nozzle (Upper) shell to Inter. shell circ. weld ~~seam~~	~~0.35~~0.10	0.0345	~~70~~101	~~30~~14%	~~49~~87	-	-	Y
Inter. shell long. weld	0.02	1.53	158	21%	125	-	-	N
Lower shell long. weld	0.02	3.86	158	27%	115	-	-	N
Inter. to lower circ. weld	0.02	2.84	100	25%	75	-	-	N
Lower shell to lower head torus circ. ~~Weld~~weld	~~0.35~~0.10	0.0341	~~70~~84	~~30~~14%	~~49~~72	-	-	Y
Lower head torus long. weld	0.02	≤ 0.0341	137	≤ 11%	≥ 122	-	-	Y
Lower head torus to dome circ. weld	0.1	< 0.0341	101	< 14%	> 87	-	-	Y

1 USE normal to the principal working direction (NPWD).
2 USE in the principal working direction (PWD).
3 Per the Weld Inspection Forms, the Unit 1 inlet nozzle to shell circumferential weld 105-121A, was fabricated using manual E-8018 type welds. The initial properties (RT_{NDT} and USE) for the E-8018 type welds are bounded by the generic Linde 0091 flux type weld properties. Therefore, the generic Linde 0091 flux type initial RT_{NDT} and USE values are used for this weld.

Table 4.2-5 STP Unit 2 Reactor Vessel Material C_V USE

Material	Chemistry		54 EFPY C_V USE at the Clad / Base Metal Interface					Ext. Beltline (Y/N)
Component	Cu (%)	Fluence (10^{19} n/cm^2)	NPWD(1) (ft - lb)	% Decrease in C_V USE	EOL NPWD (ft - lb)	PWD(2) (ft - lb)	EOL PWD (ft - lb)	
Base Metals								
Inlet Nozzle R2011-1	0.1	0.0328	165	11%	147	-	-	Y
Inlet Nozzle R2011-2	0.1	0.0328	136	11%	121	-	-	Y
Inlet Nozzle R2011-3	0.12	0.0328	128	13%	111	-	-	Y
Inlet Nozzle R2011-4	0.11	0.0328	132	12%	116	-	-	Y
Outlet Nozzle R2012-1	0.35	0.0328	132	26%	98	-	-	Y
Outlet Nozzle R2012-2	0.35	0.0328	132	26%	98	-	-	Y
Outlet Nozzle R2012-3	0.35	0.0328	121	26%	90	-	-	Y
Outlet Nozzle R2012-4	0.35	0.0328	126	26%	93	-	-	Y
Nozzle Shell R2505-1	0.05	0.0328	114	26%	84	-	-	Y
Nozzle Shell R2505-2	0.07	0.0328	124	11%	110	-	-	Y
Nozzle Shell R2505-3	0.05	0.0328	127	11%	113	-	-	Y
Inter. Shell R2507-1	0.04	2.86	109	24%	83	137	101	N
Inter. Shell R2507-2	0.05	2.86	129	24%	98	145	107	N
Inter. Shell R2507-3	0.05	2.86	122	24%	93	149	110	N
Lower Shell R3022-1	0.03	3.72	124	26%	92	141	104	N
Lower Shell R3022-2	0.04	3.72	118	26%	87	141	104	N
Lower Shell R3022-3	0.04	3.72	123	26%	91	126	93	N
Bottom Head Torus R3020-1	0.11	0.0343	86	12%	76	-	-	Y
Bottom Head Dome R3021-1	0.09	0.0343	132	11%	117	-	-	Y

Table 4.2-5 STP Unit 2 Reactor Vessel Material C_V USE

Material	Chemistry		54 EFPY C_V USE at the Clad / Base Metal Interface					Ext. Beltline (Y/N)
Component	Cu (%)	Fluence (10^{19} n/cm^2)	NPWD[1] (ft - lb)	% Decrease in C_V USE	EOL NPWD (ft - lb)	PWD[2] (ft - lb)	EOL PWD (ft - lb)	
Welds								
Inlet / Outlet nozzle to shell welds	0.10	< 0.0328	84	< 14%	> 72	-	-	Y
Nozzle (Upper) shell long. welds	0.10	≤ 0.0328	101	≤ 14%	≥ 87	-	-	Y
Nozzle (Upper) shell to Inter. shell circ. weld ~~seam~~	~~0.35~~0.10	0.0328	~~70~~84	~~60~~14%	~~49~~72	-	-	Y
Inter. shell long. weld G3.02	0.04	1.68	146	22%	114	-	-	N
Lower shell long. weld E3.12	0.04	3.72	101	26%	75	-	-	N
Inter. to lower circ. weld seam E3.12	0.04	2.84	101	24%	77	-	-	N
Lower shell to lower head torus circ. ~~Weld~~weld	~~0.35~~0.10	0.0343	~~70~~84	~~30~~14%	~~49~~72	-	-	Y
Lower head torus long. weld	0.025	≤ 0.0343	151	≤ 11%	≥ 134	-	-	Y
Lower head torus to dome circ. weld	0.1	< 0.0343	84	< 14%	> 72	-	-	Y

1 USE normal to the principal working direction (NPWD).
2 USE in the principal working direction (PWD).

Table 3.1.2-1 Reactor Vessel, Internals, and Reactor Coolant System – Summary of Aging Management Evaluation – Reactor Vessel and Internals

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Vol. 2 Item	Table 1 Item	Notes
RV Inlet and Outlet Nozzles	PB	Carbon Steel with Stainless Steel Cladding	Reactor Coolant (Int)	Cumulative fatigue damage	Time-Limited Aging Analysis evaluated for the period of extended operation	IV.A2-21	3.1.1.09	A
RV Inlet and Outlet Nozzles	PB	Carbon Steel with Stainless Steel Cladding	Reactor Coolant (Int)	Crack growth	Time-Limited Aging Analysis evaluated for the period of extended operation	IV.A2-22	3.1.1.21	A
RV Internal Disconnect Device Housing	PB	Stainless Steel	Borated Water Leakage (Ext)	None	None	IV.E-3	3.1.1.86	A
RV Upper, Intermediate, Lower Shell and Welds	PB	Carbon Steel with Stainless Steel Cladding	Reactor Coolant (Int)	Cumulative fatigue damage	Time-Limited Aging Analysis evaluated for the period of extended operation	IV.A2-21	3.1.1.09	A
RV Upper, Intermediate, Lower Shell and Welds	PB	Carbon Steel with Stainless Steel Cladding	Reactor Coolant (Int)	Crack growth	Time-Limited Aging Analysis evaluated for the period of extended operation	IV.A2-22	3.1.1.21	A

RV Upper, Intermediate, Lower Shell and Welds	PB	Carbon Steel with Stainless Steel Cladding	Reactor Coolant (Int)	Loss of fracture toughness	Time-Limited Aging Analysis evaluated for the period of extended operation	IV.A2-23	3.1.1.17	A

Enclosure 3

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
42	Enhance the Reactor Head Closure Studs program procedures 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.	B2.1.3	Starting with the current (Third Interval) 10-year ASME Section XI inspection interval CR 12-15170