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

February 16, 2012
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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 11 (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 January 30, 2012, "Requests for Additional Information for the Review of the South Texas Project, Units 1 and 2, License Renewal Application – Aging Management, Set 11 (TAC Nos. ME4936 and ME 4937)" (ML12030A164)

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.

There are no 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.

A147
NRR

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

Executed on 2/16/2012
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

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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, SET 11
(TAC NOS. ME4936 AND ME4937)

Metal Fatigue -South Texas (060)

RAI 4.3-1a (follow-up)

Background

In response to RAI 4.3-1 dated November 21, 2011, the applicant clarified that the charging flow step decrease and return to normal transient assumes 24,000 occurrences for the design number of cycles. In addition, the Fatigue Monitoring Program does not specifically count this event because the number of assumed cycles (24,000) is far greater than the number expected over 60 years.

Issue

It is not clear to the staff what the expected number of cycles is over 60 years for the charging flow step decrease and return to normal transient.

If this transient was used as an input into a fatigue time-limited aging analysis (TLAA), it is not clear to the staff why this transient does not need to be monitored by the Metal Fatigue of Reactor Coolant Pressure Boundary Program to ensure the analysis remains valid.

Request:

- Clarify the baseline number of events up to the end of 2008 and the 60-year projected cycles for the charging flow step decrease and return to normal transient. Based on the 40-year and 60-year cycles, justify how they support the statement in the response, "the' assumed cycles (24,000) are far greater than the number expected over 60 years."

- In lieu of a justification, include the letdown flow 50 percent decrease and return transient as part of the Metal Fatigue of Reactor Coolant Pressure Boundary Program.

STPNOC Response:

1) The charging flow 50 percent step decrease and return to normal transient is not included in the baseline because the transient is not monitored. This transient occurs when there is a power change, typically during plant heatup and cooldown. The estimated number of events based on the plant heatups and cooldowns that have occurred up to the year ending of 2008 are 87 (Unit 1) and 55 (Unit 2). The 60-year projected events are estimated to be 172 (Unit 1) and 154 (Unit 2). The 60-year projection is approximately one percent of the program limit value of 14,400. Note that the Norm/Alt Charge limit value of 24,000 is

revised to 14,400 in STPNOC response (Reference: NOC-AE-11002742, ML11335A131) to RAI 4.3-2. The program limit value (14,400) is far greater than the number of projected (<200) events.

The margin between 60-year projected events (<200) and program limit value (14,400) is sufficient to allow for unanticipated shutdowns or power reductions and provides reasonable assurance that the Norm/Alt Charge limit (14,400) will not be reached during the 60-year plant life. Therefore, the charging flow 50 percent step decrease and return to normal transient does not require monitoring by the Metal Fatigue of Reactor Coolant Pressure Boundary Program to ensure the analysis remains valid.

2) See justification provided in STPNOC response to item 1 above.

RAI 4.3-2a (follow-up)

Background:

In response to RAI 4.3-2 dated November 21, 2011, the applicant stated the letdown flow 50 percent decrease and return transient was included in normal and alternate charging line fatigue analyses and is not a normal operating event with the plant at power. The applicant clarified that this transient was included for conservatism and assumed to occur approximately once a week for 40 years. The number experienced will not approach the limit given the conservatism of this assumption; therefore, this transient is not counted in the Metal Fatigue of Reactor Coolant Pressure Boundary Program.

The staff noted that, as part of the response, the applicant provided a table of the transients used in the fatigue analyses to determine the break locations, in which 1200 cycles of the letdown flow 50 percent decrease and return transient were included in the normal and alternate charging line fatigue analyses.

Issue:

It is not clear to the staff what the expected number of cycles is over 60 years for the letdown flow 50 percent decrease and return transient.

If this transient was used as an input into a fatigue TLAA, it is not clear to the staff why this transient does not need to be monitored by the Metal Fatigue of Reactor Coolant Pressure Boundary Program to ensure the analysis remains valid.

Request:

- Clarify the baseline number of events up to the end of 2008 and the 60-year projected cycles for the letdown flow 50 percent decrease and return transient. Based on the 40-year and 60-year cycles, justify how it supports the statement in the response that, "the number experienced will not approach the limit given the conservatism of this assumption; therefore, this transient is not counted in the Metal Fatigue of Reactor Coolant Pressure Boundary Program."

- In lieu of a justification, include the letdown flow 50 percent decrease and return transient as part of the Metal Fatigue of Reactor Coolant Pressure Boundary Program.

STPNOC Response:

1) A typo was made in the table provided as part of the letter (Reference: NOC-AE-11002742, ML11335A131) response to RAI 4.3-2 dated November 21, 2011. The transient description of Letdown Flow 50% Decrease and Return should read letdown flow 70 percent decrease and return.

   The clarification provided in response to RAI 4.3-2 dated November 21, 2011, "this transient was included for conservatism and assumed to occur approximately once a week for 40 years", Is meant to provide the explanation for the letdown flow 70 percent decrease and return transient analyzed value of 2000 (50 x 40), not the number of projected events. As part of the response, this analyzed value of 2000 was reduced to 1200 (0.60 x 2000) as the limit value for the alternate and charging nozzles.

   The letdown flow 70 percent decrease and return to normal transient is not included in the baseline because this transient is not expected to occur. STP operates with continuous letdown at nominal flow. Letdown flow reduction is not part of normal operating practices. Since this transient is not expected to occur, the 60-year projected events are estimated to be zero (Units 1 and 2).

   The margin between 60-year projected events (zero) and Norm/Alt Charge limit value (1200) provides reasonable assurance that the program limit (1200) will not be reached during the 60-year plant life. Therefore, letdown flow 70 percent decrease and return to normal transient does not require monitoring by the Metal Fatigue of Reactor Coolant Pressure Boundary Program to ensure the analysis remains valid.

2) See justification provided in STPNOC response to item 1 above. Monitoring of the letdown flow 70 percent decrease and return to normal transient is not required.

RAI 4.3-5a (follow-up)

Background:

In response to RAI 4.3-5 dated November 21, 2011, the applicant stated for the Unit 1 Class 3 Feedwater Control Valves, the cycle limiting value remains at 10,300, as described in LRA Section 4.3.1.12.

The staff noted that LRA Section 4.3.2.12 states the following (emphasis added):

> To obtain acceptable fatigue limits the number of loadings and unloadings between 15 and 100 percent power had to be reduced from 13,200 to 10,300, of loading or unloading for Unit 2. **This limit does not apply to design of the Unit 1 feedwater control valves.**

Issue:

It appears that the applicant incorrectly referenced LRA Section 4.3.1.12 in its response, as Section 4.3.1.12 does not exist in the LRA. In addition, the applicant's statements in response to RAI 4.3-5 are not consistent with the information provided in LRA Section 4.3.2.12

Request:

- Clarify the reference to LRA Section 4.3.1.12 that was cited in response to RAI 4.3-5.
- Clarify the discrepancy between the response to RAI 4.3-5 and the information provided in LRA Section 4.3.2.12 for the limit of the number of loadings and unloadings between 15 and 100 percent power for Unit 1. Confirm that the 10,300 cycle limit for loadings and unloadings between 15 and 100 percent power is applicable to the Unit 1 feedwater control valves. Provide the necessary revisions to the response to RAI 4.3-5 and the LRA.

STPNOC Response:

1) The reference to LRA Section 4.3.1.12 cited in response to RAI 4.3-5 dated November 21, 2011 should read 4.3.2.12.

2) As stated in the response to RAI 4.3-5 dated November 21, 2011, the 10,300 cycle limit for loadings and unloadings between 15 and 100 percent power is applicable to the Unit 1 feedwater control valves.

The statement in LRA Section 4.3.2.12, that the cycle limiting value of 10,300 does not apply to design of the Unit 1 feedwater control valves, was intended to note that the number of Unit 1 loading and unloading events between 15 and 100 percent power is limited to 3000 because of the reactor vessel bottom mounted instrumentation half-nozzle repairs. The Unit 1 feedwater control valves are qualified for 10,300 events by analysis. LRA Section 4.3.2.12 is revised to clarify that the 10,300 cycle limiting value applies to both the Unit 1 and Unit 2 feedwater control valves.

Enclosure 2 provides the line-in/line-out revision to LRA Section 4.3.2.12.

RAI 4.3-8a (follow-up)

Background:

In response to RAI 4.3-5 dated November 21, 2011, the applicant stated that "the stress pairing that contributes the most to fatigue was analyzed with 13,177 events when only 10 events were required." In addition, the response states that "the Metal Fatigue of Reactor Coolant Pressure Boundary Program will maintain this margin for the original analysis during the period of extended operation by ensuring that the specified 10 events are not exceeded."

Issue:

It is not clear to the staff what "event" was analyzed for 13,177 cycles and what document (e.g. design specification, Code or Standard) required only 10 of these events to be analyzed. It is also not clear which transient is being monitored by the Metal Fatigue of Reactor Coolant Pressure Boundary Program for the "specified 10 events."

The staff reviewed LRA Table 4.3-2 and it is not clear which transient is being monitored in the Metal Fatigue of Reactor Coolant Pressure Boundary Program during the period of extended operation to ensure "that the specified 10 events are not exceeded."

Request:

- Clarify the "event" that is being referenced in the response to RAI 4.3-8 that was analyzed for 13,177 cycles.
- Clarify the requirement that specified that only 10 of these "events" had to be analyzed. Reference any applicable design specification, Code or Standard that provides this "requirement."
- Clarify the transient that is being managed by the Metal Fatigue of Reactor Coolant Pressure Boundary Program. Confirm that the "program limiting value" for this transient is 10 cycles and is incorporated into the implementing procedures for this program.

STPNOC Response:

1) The primary side hydrostatic test is the "event" referenced in the response to RAI 4.3-8. The stud hole inserts fatigue analysis pairs primary side hydrostatic test events (10) with 13,177 of the 13,200 unit unloading at 5 % of full power per minute events.

2) The stud hole inserts maximum usage factor (0.8852) is calculated using the transients listed in the table below. These transients are specified in the reactor pressure vessel design specification and are equal to or higher than the same transient listed in UFSAR Table 3.9-8.

3) The Metal Fatigue of Reactor Coolant Pressure Boundary Program (B3.1) manages the applicable transients used in the stud hole inserts fatigue analysis. Transients 5, 6, 13, 14, and 32 are not managed. The STP units are operated as base load plants; therefore, transients 5 and 6 are not managed. STP is not licensed to operate with a loop out of service; therefore, transients 13, 14, and 32 are not managed. The managed transients are included in the draft implementing procedure for this program.

Transient Description	Analyzed Cycles	Managed by B3.1
1. RCS Heatup at 100 °F/hr	200	Yes
2. RCS Cooldown at 100 °F/hr	200	Yes
5. Unit Loading at 5% of Full Power/min	13,200	No (See Note 3)
6. Unit Unloading at 5% of Full Power/min	13,200	No (See Note 3)
9. Large Step Load Decrease With Steam Dump	200	Yes

Transient Description	Analyzed Cycles	Managed by B3.1
12. Feed Water Cycle at Hot Shutdown	2000	Yes
13. Loop Out of Service, Normal (Inactive) Loop Shutdown	80	No (See Note 4)
14. Loop Out of Service, Normal (Inactive) Loop Shutdown	70	No (See Note 4)
19. & 20 Hot Hydrostatic Test	280 (See Note 1)	Yes
24. Loss of Load (Without Immediate Reactor Trip)	80	Yes
25. Loss of Power	40	Yes
26. Partial Loss of RCS Flow (Loss of One RCP)	80	Yes
27. Reactor Trip from Full Power, without Cooldown	230	Yes
28. Reactor Trip from Full Power, with Cooldown, without Safety Injection	160	Yes
29. Reactor Trip from Full Power, with Cooldown, with Safety Injection	10	Yes
30. Inadvertent RCS Depressurization	20	Yes
32. Inadvertent Startup of an Inactive RCS Loop	10	No (See Note 4)
33. Control Rod Drop	80	Yes
36. Excessive Feedwater Flow	30	Yes
43. Primary Side Hydrostatic Test	10	Yes
48. High Head Safety Injection	60	Yes
Bolt-up (See note 2)	107	Yes

Table Notes:

1) The hot hydrostatic test transient is the primary side leak test (200) and secondary side leak test (80) combined.
2) The bolt-up transient is managed by counting the 80 refueling transients listed in LRA Table 4.3-2.
3) The STP units are operated as base load plants; therefore, this transient is not expected to occur.
4) STP is not licensed to operate with a loop out of service.

RAI for elastomers exposed to lube oil

STP RAI 3.3.2.3.28-1 (079)

Background:

In LRA Table 3.3.2-28, the applicant stated that for elastomer flexible hoses exposed to a lubricating oil internal environment there is no aging effect and no AMP is proposed. The AMR line items cite generic note G. The GALL Report does not address elastomeric materials exposed to lubricating oil.

Issue:

Given that certain elastomers such as natural rubbers and ethylene-propylene-diene are not resistant to lubricating oil, the staff needs to know the material of construction of the flexible hoses to determine if there are no aging effects.

Request:

State the materials of construction of the flexible hoses exposed to lubricating oil as listed in LRA Table 3.3.2-28. If the flexible hoses are constructed of a material that is not resistant to lubricating oil, propose an aging management program or state the basis for why no aging management program is necessary.

STPNOC Response:

A review of plant documentation for the flexible hose component did not identify the specific elastomeric material. A walkdown determined that the installed flexible hoses have no serial numbers or material identification markings on them. As a result, the elastomeric material of construction is not known.

Because the elastomeric material of construction is not known, LRA Table 3.3.2-28 is revised to manage the elastomeric internal surfaces of the flexible hoses in a lube oil environment for "hardening and loss of strength" using Aging Management Program B2.1.22, Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components.

Enclosure 2 provides the line-in/line-out revision to LRA Table 3.3.2-28.

Enclosure 2

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

List of Revised LRA Sections

RAI	Affected LRA Section
RAI 4.3-5a	Section 4.3.2.12
RAI 3.3.2.3.28-1	Table 3.3.2-28

4.3.2.12 Class 1 Design of Class 3 Feedwater Control Valves

Summary Description

The STP feedwater control valves were purchased as ASME III, Class 3 valves. UFSAR Table 3.2.B-1 identifies the safety class as non-nuclear safety (NNS). Neither of these classifications indicates a TLAA. However UFSAR Table 3.9-8 associates a limiting number of occurrences of unit loading and unloading at 5% of full power for these valves, and the methods and acceptance criteria for the evaluation of the valves for these occurrences were based on Class 1 methods of paragraph NB-3545 of ASME III, 1977 Edition through the Winter 1978 Addenda.

Because the current licensing basis indicates a lifetime limit on the feedwater control valves, the analysis which supports their design is a TLAA.

Analysis

Westinghouse Equipment Specifications require that these valves be designed to transients consistent with the STP 40-year design. However, as a result of the STP replacement steam generator project, the main feedwater control valves were analyzed for a new set of operating design transient conditions, and it was found that they could not be qualified for the full number of loading and unloading transients defined for the life of the plant. To obtain acceptable fatigue limits the number of loadings and unloadings between 15 and 100 percent power had to be reduced from 13,200 to 10,300. ~~, of loading or unloading for Unit 2. This limit does not apply to design of the Unit 1 feedwater control valves.~~

Loading and unloading events are the largest contributor to fatigue in the feedwater control valves. All other transients contribute 0.055 to the 40-year CUF. The STP units do not operate in a load-following mode, and therefore the expected number of occurrences is only a small fraction of the design number of occurrences.

STP has experienced 62 occurrences of this transient for Unit 1 and 43 occurrences for Unit 2 through July 27, 1989, less than 17% of the 385 anticipated at that point in the design life. That ratio applied to the design number for 40 years, 17% of 13,200, is 2,244 occurrences. This value can be extrapolated to 60 years by multiplying it by 1.5 (60/40), resulting in 3,366 events over 60 years. This demonstrates a large margin between the analyzed value, 10,300, and the number projected, 3,366; thus t he analysis is valid for the period of extended operation.

Disposition: Validation, 10 CFR 54.21(c)(1)(i)

The STP units do not operate in a load following mode and therefore the expected number of loading and unloading occurrences is only a small fraction of the design number of occurrences, resulting in a large margin between the analyzed value, 10,300 cycles, and the number projected, 3,366 cycles. Therefore the fatigue analysis for the STP feedwater control valves is valid for the period of extended operation. This TLAA is dispositioned in accordance with 10 CFR 54.21(c)(1)(i).

Table 3.3.2-28 Auxiliary Systems – Summary of Aging Management Evaluation – Lighting Diesel Generator

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG 1801 Vol 2 Item	Table 1 Item	Notes
Flame Arrestor	PB	Carbon Steel	Plant Indoor Air (Ext)	Loss of material	External Surfaces Monitoring Program (B2.1.20)	VII.I-8	3.3.1.58	B
Flexible Hoses	PB	Elastomer	Lubricating Oil (Int)	~~None~~ Hardening and loss of strength	~~None~~ Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (B2.1.22)	~~None~~	~~None~~	~~G~~ F, 2
Flexible Hoses	PB	Elastomer	Plant Indoor Air (Ext)	Hardening and loss of strength	External Surfaces Monitoring Program (B2.1.20)	VII.F1-7	3.3.1.11	E

Notes for Table 3.3.2-28

Standard Notes

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

B Consistent with NUREG-1801 item for component, material, environment, and aging effect. AMP takes some exceptions to 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.

F Material not in NUREG-1801 for this component.

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 This non NUREG-1801 line item was created because there is no line item for a component made of elastomer with a lubricating oil internal environment.