STP Nuclear Operating Company

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

September 15, 2011
NOC-AE-11002731
10CFR54
STI: 32913936
File: G25

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 (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 August 15, 2011, "Requests for Additional Information for the Review of the South Texas Project, Units 1 and 2 License Renewal Application – Aging Management Programs Audit, Reactor Systems" (ML11214A027)

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 requests for additional information is provided in the Enclosure to this letter.

Enclosure 2 to this letter contains one new regulatory commitment.

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 9-15-2011
Date

G. T. Powell

G. T. Powell
Vice President,
Technical Support & Oversight

KJT

Enclosures:
1. STPNOC Response to Requests for Additional Information
2. New Regulatory Commitment

A147
NRR

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Enclosure 1

STPNOC Response to Requests for Additional Information

SOUTH TEXAS PROJECT, UNITS 1 AND 2
LICENSE RENEWAL APPLICATION
REQUESTS FOR ADDITIONAL INFORMATION
AGING MANAGEMENT PROGRAMS AUDIT,
REACTOR SYSTEMS

Note: In all cases unless otherwise noted, references to the generic aging lessons learned GALL Report, a GALL aging management program (AMP), or the SRP-LR refer to the current approved revision, Revision 2.

RAI B2.1.3-1

Background:

License renewal application (LRA) Section B2.1.3 describes the applicant's Reactor Head Closure Studs Program and indicates that Regulatory Guide 1.65 (issued in October 1973) states that the ultimate tensile strength of the stud bolting material should not exceed 170 ksi. LRA Section B2.1.3 also states that one closure head insert has a tensile strength of 174.5 ksi, and identifies the use of the closure head insert as an exception that affects the "scope of program" program element. The LRA further states that the applicant credits inservice inspections that are within the scope of this AMP, which are implemented in accordance with the Inservice Inspection Program, Examination Category B-G-1 requirements, as the basis for managing cracking in these components.

In comparison, GALL AMP XI.M3, "Reactor Head Closure Stud Bolting," references the guidance outlined in RG 1.65, Rev. 1, "Materials and Inspections for Reactor Vessel Closure Studs," issued in April, 2010. RG 1.65, Rev. 1 and the GALL Report recommend using bolting materials that have a measured yield strength not exceeding 150 ksi in order to ensure the material's resistance to stress corrosion cracking (SCC).

In its review, the U.S. Nuclear Regulatory Commission (NRC or the staff) noted that STP UFSAR Tables 5.3-5 and 5.3-6 describe the reactor vessel fastener material properties of STP Units 1 and 2, respectively, including the yield strength data of the reactor head closure stud bolting material. The staff also noted that several bars of the closure stud bolting material have yield strength levels greater than 150 ksi and up to 158 ksi.

Issue:

In contrast with GALL AMP XI.M3 and RG 1.65, Rev. 1, LRA Section B2.1.3 does not clearly address a provision that precludes use of stud bolting materials with a measured yield strength level greater than 150 ksi, or justification for the use of the high-strength material. The staff also found a need to clarify how the applicant's program considers and evaluates the yield strength levels of reactor head closure stud bolting materials to adequately manage SCC.

Request:

1. 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. In addition, describe whether or not the STP Nuclear Operating Company's (the applicant) program has a provision to ensure no use of closure stud bolting materials with measured yield strength greater than 150 ksi.

   If the program does not have such a provision, further justify the adequacy of the applicant's program to manage cracking due to SCC of the high-strength material. As part of the justification, describe (a) whether or not the operating experience indicates that the closure stud bolting has been exposed to reactor coolant leakage, and (b) how the applicant's program manages the potential exposure of closure stud bolting to borated water and potential contamination that may facilitate stress corrosion cracking of the reactor head closure stud bolting components.

2. Describe whether or not the applicant's program precludes future additions of reactor head closure stud bolting components with yield strength exceeding 150 ksi to the existing set of the closure stud bolting components that are currently used in STP Units 1 and 2.

3. Revise the LRA to be consistent with the response to this RAI.

STPNOC Response:

1. Several components in the reactor vessel closure head stud assemblies have a measured yield strength greater than or equal to 150 ksi. 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, although stud assemblies exceeding this criterion will be allowed providing they are currently in use or are spare components currently on site.

   The program to manage cracking and loss of material in the reactor vessel closure head stud assemblies includes the following elements:

   - The reactor vessel closure stud assemblies are fabricated of SA-540 Grade B-24, which is included in the list of acceptable materials in Regulatory Guide 1.65, Revision 1.
   - The reactor vessel closure head stud assemblies are not metal-plated.
   - A manganese phosphate surface treatment was used when the stud assemblies were manufactured.
   - Visual and volumetric examinations are performed in accordance with ASME Section XI, Subsection IWB requirements and as recommended in Regulatory Guide 1.65.
   - Procedures require the studs, nuts, and washers to be removed and placed in storage racks during preparation for refueling to prevent exposure to the borated refueling cavity water. The stud holes in the reactor flange are sealed with special plugs, thus preventing leakage of the borated refueling water into the stud holes.
   - Neolube and 2001 Penetrating Space Age Oil are used as lubricants on stud assemblies after cleaning and examinations are complete.

   Based on the above information, the current program adequately manages potential stress corrosion cracking of the reactor head closure stud assemblies. To date, the reactor vessel head stud assemblies have not been exposed to borated reactor coolant leakage. There have been no cases of cracking of these components.

2. 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. Use of installed components and any spare components currently on site will be allowed. Allowing future use of the existing spare reactor head closure stud assemblies is justified based on plant-specific operating experience and the aging management program discussed above. South Texas Project (STP) has not experienced SCC of the reactor head closure stud assemblies. The existing spare stud assemblies, if used as a future replacement, would experience less than 40 years of service to the end of the period of extended operation. There are five spare reactor vessel closure stud assemblies on site.

3. LRA Appendix B2.1.3 and LRA Basis Document XI.M3, 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.

RAI B2.1.3-2

Background:

SRP-LR Section A.1.2.3.10 states that the operating experience of AMPs that are existing programs, including past corrective actions resulting in program enhancements or additional programs, should be considered, and that past failure would not necessarily invalidate an AMP because the feedback from operating experience should have resulted in appropriate program enhancements or new programs. The SRP-LR also states that this information should provide objective evidence to support the conclusion that the effects of aging will be managed adequately so that the structure-and component-intended function(s) will be maintained during the period of extended operation.

In its review of the applicant's operating experience related to the Reactor Head Closure Studs Program during the audit, the staff noted that a work order dated April 12, 2007, indicates that an American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI replacement of the #30 ROTO-LOK stud was conducted in Unit 2 during Refueling Outage 12 per the disposition of a design change package dated April 9, 2007. The design change package indicates that Stud #30 of Unit 2 had rotated inadvertently during the detensioning process, causing it to partially engage inside the stud insert, which is also called bushing, and this condition caused damage to all of the lugs of the stud that were partially engaged. The design change package also indicates that the applicant decided that Stud #30 of Unit 2 was to be replaced by a spare stud of the same kind from the warehouse. Based on the evaluation performed on the stud insert, the applicant determined that the non-conforming condition of the stud insert is dispositioned as "Use-As-Is." The applicant's design change package further indicates that the damaged areas of the insert lug bearing surfaces were conservatively estimated to be 17 percent of the original areas of contact.

In its review during the audit, the staff noted that the applicant's inservice inspection plan, Rev. 4, dated September 29, 2008, specifies the inspection plan for the second interval that started from September 2000 and October 2000 for Units 1 and 2, respectively. The staff also noted that Examination Category B-G-1, Item No. 86.50 in the applicant's inspection plan indicates that alternative volumetric examination is specified as the inspection method for the closure bushings, which are also called stud inserts, instead of visual VT-1 examination specified in Table IWB-2500-1 of the 2004 edition of the ASME Code, Section XI, Subsection IWB. The staff further noted that LRA Section B2.1.3 does not identify this alternative volumetric examination as an exception of the program.

Issue:

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 stress corrosion cracking may be facilitated. In addition, the partially damaged stud insert may cause partial engagement and galling of the stud bolting, and an adverse effect on the prevention of reactor vessel flange leakage. Therefore, the staff found a need to confirm why no replacement of the partially damaged stud insert is acceptable to manage loss of material.

The staff also finds that visual VT-1 examination of closure bushings is effective to detect, monitor, and manage loss of material due to wear or corrosion, and to identify and monitor a change in the condition of the damaged stud insert, especially a reduction in the load bearing surfaces. Therefore, the staff found a need to confirm whether or not the alternative volumetric examination of the closure bushings without VT-1 examination specified in ASME Code, Section XI is adequate to manage the aging effects of the closure stud inserts.

In addition, the staff found a need to clarify why the alternative volumetric examination of the stud inserts is not an exception of the applicant's program and whether or not the applicant's operating experience supports the applicant's conclusion that the program is adequate to manage the aging effects.

Request:

1. Describe whether or not a reactor head closure stud, stud insert or reactor vessel flange surface has experienced corrosion or stress corrosion cracking due to reactor vessel flange leakage, or other contact with borated water.

2. Justify why the alternative volumetric examination of the stud inserts is not an exception of the applicant's program.

3. In view that the partially damaged stud insert has not been replaced, if existent, describe the results of inspection activities that the applicant has conducted to monitor any change in the affected load bearing areas of the partially damaged stud insert.

4. As addressed above, 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 stress corrosion cracking may be facilitated. The partially damaged stud insert may cause partial engagement and galling of the stud bolting, and may be an adverse effect on the prevention of reactor vessel flange leakage.

   In view of these potential adverse effects, justify why no replacement of the partially damaged stud insert is acceptable to manage loss of material and cracking. In addition, in view that a VT-1 examination is effective to identify and monitor a reduction in the load bearing surfaces, justify why the alternative volumetric examination, without VT-1 examination specified in ASME Code, Section XI, is adequate to manage loss of material and cracking.

5. Based on the information and evaluation addressed above, if items, such as replacement of the damaged stud insert and/or augmented inspection, are identified to be added to the applicant's aging management program, describe the items and applicant's commitments associated with them, including the implementation schedules.

STPNOC Response:

1. South Texas Project (STP) reactor head closure studs have not been exposed to borated water from reactor vessel flange leakage. During refueling when the refueling cavity is flooded, plugs are inserted into the stud holes in the reactor vessel flange to prevent borated water from coming in contact with the stud hole inserts. There have been instances where the stud hole plugs have leaked, exposing the inserts to borated water. When a stud hole plug is discovered to have leaked, the stud holes are cleaned. Periodic inspections of the stud hole inserts are performed as required by ASME Code Section XI, Table IWB-2500-1 to verify their integrity. No corrosion has been found on the closure studs and inserts.

   The reactor vessel flange is regularly exposed to borated water when the refueling cavity is flooded for refueling. A thin layer of corrosion forms on the flange while the refueling cavity is flooded, but this is removed when the flange is cleaned after the refueling cavity is drained.

   No stress corrosion cracking has been found on the closure studs, inserts, or reactor vessel flange.

2. The bushing inspections will be performed in accordance with the ASME Code Section XI, Table IWB-2500-1, as required by the GALL. Table 2500-1 specifies VT-1 examination of reactor vessel pressure retaining bushings. For the current inspection interval, the NRC approved a relief request allowing ultrasonic examination of the bushings in lieu of the VT-1 examination. The safety evaluation (ML110840076) of the approved relief request stated that the ultrasonic examination is equivalent to the VT-1 examination. Inspections performed during the period of extended operation will be in accordance with the applicable Code edition. If any variances from the Code requirements are required, the variances will be submitted for approval through the relief request process.

3. An analysis was performed of the damaged stud hole insert. The damaged areas of the insert lug bearing surfaces were conservatively estimated to be 17 percent of the original areas of contact. The analysis assumed that load would not be transferred to any of the damaged insert lug bearing surfaces. Based on this loss of bearing area, the bearing stress is still acceptable and the stud hole insert is accepted for "Use-As-Is". No tests other than those that are used during normal stud installation are required.

4. As described above, an analysis of the damaged stud hole insert determined that even with a loss of 17 percent of the bearing surface area, the bearing stress is acceptable so that it is not necessary to replace the insert. The ultrasonic examination method was demonstrated in 1998 using a calibration block prepared from an actual spare stud hole insert. The calibration block included notches at different depths on both the inside and outside surfaces to ensure examination volume coverage and sensitivity were obtained. The notches were representative of flaws that would be found in-service. The demonstration confirmed that all inside and outside surface indications could be easily observed. This is superior to a VT-1 examination since portions of the lugs are not accessible with the inserts in place.

5. No LRA or AMP changes are required.

RAI B2.1.3-3

Background:

LRA Section B2.1.3 describes the applicant's Reactor Head Closure Studs Program. LRA Section B2.1.3 also states that the program manages cracking and loss of material by conducting ASME Code, Section XI inspections of reactor vessel flange stud hole threads, reactor head closure studs, nuts, washers, and bushings. Consistently, LRA Table 1 item 3.1.1.71, indicates that the applicant uses the Reactor Head Closure Studs Program to manage cracking and loss of material of high-strength low alloy steel closure head stud assembly exposed to air with reactor coolant leakage.

In comparison, LRA Table 3.1.2-1 includes AMR line items to manage cracking and loss of material of "RV closure head bolts"; however, LRA Table 3.1.2-1 does not clearly indicate that the AMR line items include the other reactor head closure stud bolting components such as reactor vessel flange stud hole threads, nuts, washers and bushings as addressed in LRA Section B2.1.3. Furthermore, the staff noted that the on-site documentation regarding the screening of the reactor vessel components for aging management includes only 72 reactor vessel head closure bolts, but it does not include any other reactor vessel head closure bolting component such as washers, bushings, nuts, and threads in the reactor vessel flange.

By contrast, the "scope of the program" program element of GALL AMP XI.M3, "Reactor Head Closure Stud Bolting," indicates that the program manages cracking and loss of material for reactor head closure stud bolting (studs, washers, bushings, nuts, and threads in flange) for both BWRs and PWRs. More specifically, GALL Report items IV.A2.RP-52 and IV.A2.RP-53 address the AMR line items for PWRs to manage cracking and loss of material, respectively, of the reactor vessel closure head stud assembly.

Issue:

The AMR line items addressed in LRA Table 3.1.2-1 to manage cracking and loss of material of reactor head closure stud bolting are not clear as to whether or not the line items include the closure studs, nuts, washers, bushings and flange threads, consistent with the GALL Report and LRA Section B2.1.3.

Request:

Describe whether or not the AMR line items addressed in LRA Table 3.1.2-1 to manage cracking and loss of material of reactor head closure stud bolting include the closure studs, nuts, washers, bushings and flange threads.

In addition, revise the LRA and on-site documentation consistent with the applicant's response to this RAI.

STPNOC Response:

The AMR line items for component type "RV Closure Head Bolts" in LRA Table 3.1.2-1 that manage cracking and loss of material includes the closure studs, nuts, washers, and bushings, all of which are high strength carbon steel. The flange threads are in the reactor vessel which is carbon steel. Reactor vessel flange thread inspections are part of the AMP B2.1.3, Reactor Vessel Closure Studs program.

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."

RAI B2.1.19-1

Background:

GALL AMP XI.M35 provides specific guidance regarding small-bore piping inspection sampling. Based on the applicant's plant-specific operating experience, the inspection sampling should include ten percent of the weld population or a maximum of 25 welds of each weld type (e.g., butt weld, socket weld, etc.) using a methodology to select the most susceptible and risk-significant welds.

LRA Section B2.1.19, "One-Time Inspection of ASME Code, Class 1 Small-Bore Piping," as amended by letter dated June 16, 2011, states that, for ASME Code, Class 1 small-bore piping, the ISI Program (ASME Code, Section XI Inservice Inspection, Subsections IWB, IWC and IWD) requires volumetric (ultrasonic) examinations on selected butt weld locations to detect cracking.

Issue:

The applicant did not provide specific information regarding the small bore piping weld population for either butt welds or socket welds. In addition, the applicant did not provide specific details regarding the butt weld inspection sampling size. This information is needed by the staff to evaluate the adequacy of the applicant's inspection sampling for socket welds and butt welds and whether the applicant's program is consistent with the recommendations of GALL AMP XI.M35.

Request:

Describe the total populations of Class-1 small-bore welds for each weld type (e.g., butt weld, socket weld, etc.) for each unit. Provide the inspection sample size for Class 1 small-bore butt welds in terms of number of welds and percentage of the weld population for each unit.

STPNOC Response:

LRA Appendix A1.19, Appendix B2.1.19 and LRA Basis Document AMP XI.M35 (B2.1.19), One-Time Inspection of ASME Code Class 1 Small-Bore Piping program, will be revised to include the following:

> Unit 1 has 182 Class 1 small-bore butt welds and 49 Class 1 small-bore socket welds. The inspection sample for the Unit 1 Class 1 small-bore butt welds is 19 and the inspection sample for the Unit 1 Class 1 small-bore socket welds is 5, which are 10 percent of each population. In Unit 2, there are 190 Class 1 small-bore butt welds and 59 Class 1 small-bore socket welds. The inspection sample size for the Unit 2 Class 1 small-bore butt welds is 19 and the inspection sample size for Unit 2 Class 1 small-bore socket welds is 6, which are 10 percent for each population

RAI B2.1.21-1

Background:

LRA Section B2.1.21, "Flux Thimble Tube Inspection," states that if the current measured wear exceeds the acceptance criteria or if the predicted wear (as a measure of percent through-wall) for a given flux thimble tube is projected to exceed the established acceptance criteria prior to the next refueling outage, corrective actions are taken to reposition, cap, or replace the tube. The "monitoring and trending" program element of GALL AMP XI.M37, "Flux Thimble Tube Inspection," states that flux thimble tube wall thickness measurements are trended and wear rates are calculated based on plant-specific data. In addition, it states that wall thickness is projected using plant-specific data and a methodology that includes sufficient conservatism to ensure that wall thickness acceptance criteria continue to be met during plant operation between scheduled inspections.

The "acceptance criteria" program element of GALL AMP XI.M37 states that the acceptance criteria should include allowances for factors such as instrument uncertainty, uncertainties in wear scar geometry, and other potential inaccuracies, as applicable, to the inspection methodology chosen.

Issue:

LRA Section B2.1.21 and the on-site documentation related to this program do not clearly address how the program manages the discrepancies between projected wall loss and measured wall loss. Specifically, during the audit, the applicant indicated that there was an instance that the applicant took corrective actions after the measured wall loss exceeded the acceptance criterion of 80 percent wall loss. Such instances indicate that the program may be under-predicting the amount of wear that is occurring in the tubes.

Request:

1. Provide a summary of the flux thimble tube inspection results over the last three inspection outages for each unit and identify how many times the actual wear results were non-conservative when compared to the prior trending (wear projection) basis. For each instance identified above, if applicable, identify the under-prediction of the wall loss as a percentage of the tube's nominal wall thickness.

2. Describe how the program identifies and reconciles discrepancies between projected wall loss and measured wall loss, especially for the cases in which the discrepancies are large and unexpected. Specifically, clarify how the program re-baselines and adds conservatisms in the new trending basis when the actual inspection results demonstrate that the prior trending basis was not conservative.

3. Clarify how the program accounts for instrument and wear scar uncertainties in the trending basis or acceptance criterion, consistent with the "acceptance criteria" program element recommendation in GALL AMP XI.M37.

   In addition, justify why the current wear projection methodology (i.e., trending basis) is conservative and adequate for managing loss of material due to wear in the flux thimble tubes.

STPNOC Response:

1. The following table shows measured wear, predicted wear, and the amount of under prediction of wear in percent wall thickness for the Unit 1 flux thimble tubes during the last three refueling outages when wear was measured. Based on projections from measurements from 1RE12 and 1RE13, no measurements were taken in 1RE14. Flux thimble tubes that have not shown any wear are not included in the table. ND indicates no detectable wear.

Unit 1											
	March 2005 Refueling Outage (1RE12)				**October 2006 Refueling Outage (1RE13)**				**October 2009 Refueling Outage (1RE15)**		
A	**B**	**C**	**D**		**B**	**C**	**D**		**B**	**C**	**D**
C7	NA	31	*		60	29	0		29	38	9
C8	NA	25	*		48	31	0		43	43	0
H3	NA	36	*		70	38	0		42	26	0
H13	NA	ND	NA		NA	ND	NA		NA	23	*
J8	NA	ND	NA		NA	ND	NA		NA	18	*
R11	NA	ND	NA		NA	ND	NA		NA	15	*
E5	NA	28	*		54	24	0		24	21	0
H3	NA	21	*		41	21	0		21	20	0
D8	NA	15	*		29	16	0		18	ND	0

Legend: A – Location
B – Predicted wear in percent wall thickness
C – Measured wear in percent wall thickness
D – Amount of under prediction of wear in percent wall thickness
NA – not applicable because no wear detected
ND – indicates no detectable wear
* - no under prediction is counted for initial appearance of wear

The following table shows measured wear, predicted wear, and the amount of under prediction of wear in percent wall thickness for the Unit 2 flux thimble tubes during the last three refueling outages. ND indicates no detectable wear. The flux thimble tube at location G9 was repositioned 36 inches during refueling outage 2RE11 because an 85 percent wall loss was predicted during the next refueling cycle.

Unit 2												
	April 2007 Refueling Outage (2RE12)				**October 2008 Refueling Outage (2RE13)**				**April 2010 Refueling Outage (2RE14)**			**Comment**
A	**B**	**C**	**D**		**B**	**C**	**D**		**B**	**C**	**D**	
A9	20	24	4		29	24	0		NA	ND	NA	replaced 2RE13
A9	51	19	0		19	18	0					
A9	NA	22	*		45	21	0					
A9	NA	19	*		39	17	0					
A9	51	53	2		57	52	0					
A9	39	18	0		18	16	0					
A11	43	41	0		41	40	0		NA	ND	NA	replaced 2RE13

Unit 2												
	April 2007 Refueling Outage (2RE12)				October 2008 Refueling Outage (2RE13)				April 2010 Refueling Outage (2RE14)			Comment
A	B	C	D		B	C	D		B	C	D	
A11	65	67	2		71	68	0					
A11	50	54	4		59	56	0					
C5	33	32	0		32	30	0		30	33	3	
C5	17	18	1		20	17	0		17	19	2	
C7	29	28	0		28	26	0		26	23	0	
C7	37	37	0		37	33	0		33	39	6	
C7	22	19	0		19	18	0		18	17	0	
C8	32	16	0		16	15	0		15	19	4	
C8	34	17	0		17	14	0		14	20	6	
C8	39	41	2		44	43	0		45	56	11	
D8	35	25	0		25	21	0		21	24	3	
D8	34	40	6		46	41	0		42	45	3	
D8	32	18	0		21	17	0		17	16	0	
D10	21	19	0		19	18	0		18	15	0	
D10	31	33	2		46	35	0		37	32	0	
D12	15	14	0		14	14	0		14	15	1	
D12	36	17	0		17	16	0		16	14	0	
D12	23	23	0		23	21	0		21	24	3	
E5	24	26	2		29	23	0		23	31	8	
E5	17	16	0		16	14	0		14	18	4	
E5	18	17	0		17	16	0		16	18	2	
E9	34	35	1		37	34	0		34	33	0	
E9	18	17	0		17	17	0		17	17	0	
E9	27	24	0		24	25	1		26	24	0	
E9	17	15	0		15	15	0		15	14	0	
E11	29	31	2		34	31	0		31	32	1	
F1	28	33	5		39	31	0		NA	ND	NA	replaced 2RE13
F1	36	33	0		33	31	0					
F1	46	52	6		59	55	0					
F1	61	62	1		64	59	0					
F3	31	17	0		35	17	0		17	14	0	
F3	44	30	0		30	30	0		30	31	1	
F3	40	25	0		29	23	0		23	25	2	
F3	52	40	0		40	38	0		38	40	2	
F3	31	26	0		26	35	9		44	41	0	
F3	33	36	3		40	34	0		34	42	8	
F7	29	28	0		28	30	2		30	33	3	
F7	29	32	3		36	29	0		29	31	2	
F7	40	44	4		49	43	0		43	40	0	
F7	39	47	8		56	41	0		41	43	2	
F8	19	17	0		17	16	0		NA	ND	NA	replaced 2RE13
F8	19	15	0		15	13	0					
F8	55	60	5		66	58	0					
F14	31	32	1		34	29	0		29	31	2	
F14	41	38	0		38	38	0		38	39	1	
F14	33	28	0		28	27	0		27	29	2	
F14	36	30	0		30	30	0		30	34	4	
G5	31	38	7		46	35	0		35	33	0	
G5	36	18	0		37	12	0		12	15	3	
G5	20	20	0		23	17	0		17	17	0	

Unit 2												
	April 2007 Refueling Outage (2RE12)				October 2008 Refueling Outage (2RE13)				April 2010 Refueling Outage (2RE14)			Comment
A	B	C	D		B	C	D		B	C	D	
G5	25	28	3		32	25	0		25	19	0	
G5	26	28	2		31	27	0		27	28	1	
G9	43	44	1		46	39	0		NA	ND	NA	replaced 2RE13
G9	58	50	0		50	40	0					
G9	41	42	1		44	42	0					
G9	NA	ND	NA		NA	16	*					
G9	NA	25	*		51	15	0					
G9	NA	14	*		29	11	0					
G9	50	62	12		71	58	0					
G9	38	17	0		17	12	0					
G9	35	37	2		40	25	0					
G12	29	22	0		22	20	0		20	19	0	
H3	25	28	3		32	25	0		25	29	4	
H3	38	43	5		49	31	0		31	39	8	
H3	22	19	0		19	13	0		13	20	7	
H3	26	15	0		18	12	0		12	17	5	
H4	23	15	0		15	13	0		13	18	5	
H4	36	33	0		33	31	0		31	32	1	
H6	48	54	6		61	52	0		NA	ND	NA	replaced 2RE13
H6	48	53	5		59	48	0					
H6	18	15	0		15	15	0					
H6	NA	25	*		51	15	0					
H6	21	18	0		18	12	0					
H11	24	23	0		23	24	1		25	22	0	
H11	25	30	5		36	26	0		26	29	3	
H11	36	23	0		29	20	0		20	18	0	
H13	39	42	3		46	39	0		39	42	3	
H13	38	47	9		59	47	0		47	52	5	
H13	36	29	0		29	27	0		27	31	4	
H13	29	23	0		23	19	0		19	21	2	
H15	28	31	3		35	30	0		NA	ND	NA	replaced 2RE13
H15	56	60	4		65	60	0					
H15	39	34	0		34	33	0					
H15	37	35	0		35	35	0					
H15	32	14	0		14	11	0					
J1	25	19	21		19	16	0		NA	ND	NA	replaced 2RE13
J1	44	64	20		85	67	0					
J1	NA	ND	NA		NA	23	*					
J1	NA	24	NA		49	23	0					
J1	NA	ND	NA		NA	43	*					
J1	43	52	9		62	48	0					
J7	23	18	0		18	18	0		18	21	3	
J7	31	36	5		43	38	0		40	36	0	
J7	34	17	0		17	12	0		12	15	3	
J8	19	17	0		17	14	0		14	14	0	
J8	15	15	0		15	12	0		12	13	1	
J10	25	27	2		30	25	0		25	24	0	
J10	16	17	1		19	15	0		15	16	1	

Unit 2												
	April 2007 Refueling Outage (2RE12)				October 2008 Refueling Outage (2RE13)				April 2010 Refueling Outage (2RE14)			Comment
A	**B**	**C**	**D**		**B**	**C**	**D**		**B**	**C**	**D**	
J14	27	30	3		34	22	0		NA	ND	NA	replaced 2RE13
J14	25	27	2		31	30	0					
J14	31	35	4		40	32	0					
J14	62	65	3		75	59	0					
K12	18	16	0		16	17	1		18	26	8	
K12	NA	ND	NA		NA	16	*		32	17	0	
K12	19	19	0		19	17	0		17	21	4	
K12	17	16	0		16	12	0		12	17	5	
K12	42	40	0		40	36	0		36	38	2	
L5	25	26	1		28	29	1		32	25	0	
L5	36	41	5		47	44	0		47	41	0	
L5	34	33	0		33	37	4		41	37	0	
L5	34	45	11		57	38	0		38	41	3	
L8	46	47	1		49	44	0		44	44	0	
L8	20	22	2		25	22	0		22	19	0	
L8	28	31	3		35	30	0		30	32	2	
L8	35	39	4		44	38	0		38	43	5	
L10	19	17	0		17	15	0		15	14	0	
L10	26	15	0		18	11	0		11	11	0	
L11	15	14	0		14	14	0		14	13	0	
L11	20	21	1		23	18	0		18	19	1	
M7	22	22	0		22	23	1		24	21	0	
M7	42	46	4		57	46	0		46	55	9	
M7	21	17	0		17	24	7		31	33	2	
M7	NA	18	*		37	17	0		17	21	4	
N2	19	22	3		26	17	0		NA	ND	NA	replaced 2RE13
N2	60	65	5		71	64	0					
N2	36	40	4		45	36	0					
N2	28	30	2		33	28	0					
N4	22	25	3		29	22	0		22	25	3	
N4	27	26	0		26	26	0		26	25	0	
N6	22	31	9		41	20	0		20	28	8	
N6	38	43	5		49	43	0		43	44	1	
N6	22	20	0		20	15	0		15	18	3	
N6	38	43	5		29	38	9		38	39	1	
N8	45	53	8		62	48	0		48	50	2	
N8	24	28	4		33	17	0		17	22	5	
R6	34	16	0		16	20	4		24	21	0	
R6	21	21	0		21	20	0		20	24	4	
R6	16	17	1		17	17	0		17	23	6	

<u>Legend</u>:

A – Location
B – Predicted wear in percent wall thickness
C – Measured wear in percent wall thickness
D – Amount of under prediction of wear in percent wall thickness
NA – not applicable because no wear detected
ND – indicates no detectable wear

* - no under prediction is counted for the initial appearance of wear

Notes:

a. Most flux thimble tubes for Unit 2 were found with multiple locations of wear as indicated in the table.
b. For the Unit 2 April 2010 refueling outage (2RE14), "blank cells" in the table indicate no wear measured at any location of the flux thimble tube.

2. The method used to predict future flux thimble tube wear is to use the two most recent measurements and linearly extrapolate to the time when the next measurement will be made. Westinghouse states in WCAP-12866 that flux thimble tube wear is generally non-linear with the rate decreasing with time. A comparison of flux thimble tube wear measurements and corresponding wear projections for the three most recent refueling outages indicates that measured wear is greater than the projected wear 8 percent of the time for Unit 1 and 37 percent of the time for Unit 2. Thus, linear extrapolation is a slightly conservative and reasonable method of predicting future flux thimble tube wear.

   In cases where the projected wear of a flux thimble tube is not conservative compared to the measured value, "re-baselining" is an integral part of the wear projection methodology because the linear extrapolation uses measured wear in projecting future wear. Consequently, if a wear measurement is higher than expected, the methodology for projecting wear will cause the new projection to be correspondingly higher.

3. Westinghouse reported in WCAP-12866 that, based on measurements performed in a hot cell facility, eddy current measurements were accurate or conservative, so that it is not necessary to add additional uncertainty margin to the measurements. Westinghouse also determined that flux thimble tubes have a high residual strength with wall loss on the order of 90 percent, and will retain their functional and structural integrity with up to 85 percent wall loss. For conservatism, an acceptance criterion of 80 percent wall loss has been set where action (replacement, repositioning, or removal from service) must be taken.

   As discussed above, the methodology of using linear extrapolation to predict future wear is conservative compared to the method recommended by Westinghouse in WCAP-12866. In addition, a review of historical results indicates that in the majority of cases, projections of wall loss have been higher than the measured wall loss. In most of the cases where the projections have under predicted the wall loss, the under prediction has been by only a few percent of wall loss. Using this methodology, measured flux thimble wear has not exceeded the acceptance criterion of 80 percent wall loss with the exception of three occurrences in 1997. It can thus be concluded that the methodology for predicting flux thimble wear is reasonable and conservative.

RAI B2.1.21-2

Background:

LRA Section B2.1.21 states that the applicant's Flux Thimble Tube Inspection Program is an existing program which implements the recommendations of NRC Bulletin 88-09, "Thimble Tube Thinning in Westinghouse Reactors." LRA Section B2.1.21 also describes several program enhancements regarding the creation of new program procedures.

SRP-LR Section 3.0.1 states that enhancements are revisions or additions to existing aging management programs that the applicant commits to implement prior to the period of extended operation. The SRP-LR also states that enhancements include, but are not limited to, those activities needed to ensure consistency with the GALL Report recommendations and that these enhancements may expand, but not reduce, the scope of an AMP.

Issue:

The LRA states that the program is an existing program and implements the recommendations of NRC Bulletin 88-09. However, the staff noted that the program enhancements address most of the major technical aspects of the program. Therefore, the staff found a need to clarify which portion of each enhancement is the revision or addition to the existing program. If the new program procedures include a change to the technical aspects of the existing program (e.g., a change to wear projection methodology or acceptance criteria for wall loss), the technical basis of the change to the existing program needs to be addressed.

Request:

For each enhancement, describe which portion of the enhancement is the revision or addition to the existing program. Especially, clarify which portion of each enhancement is the revision or addition to the technical aspects that have been implemented in the existing program.

In addition, if the new program procedures include a change to the technical aspects of the existing procedures (e.g., a change to wear projection methodology or acceptance criteria for wall loss), describe the technical basis of the change in order to justify why the change to the existing program is adequate to manage loss of material of the flux thimble tube.

STPNOC Response:

The existing Flux Thimble Tube Inspection program meets the technical requirements of NUREG-1801, "Generic Aging Lessons Learned Report". Enhancements to the Flux Thimble Tube Inspection program do not revise or make additions to the technical requirements of the existing program. The enhancements are captured in a plant procedure that documents the tube inspection practices.

RAI B2.1.21-3

Background:

LRA Section B2.1.21 describes the applicant's operating experience regarding its Flux Thimble Tube Inspection Program and it states that corrective actions taken in response to the results of the inspections in Unit 2 included repositioning of thimble tubes and replacing 25 thimble tubes with chrome plated tubes.

GALL AMP XI.M37, "Flux Thimble Tube Inspection," states that examination frequency is based upon actual plant-specific wear data and wear predictions that have been technically justified as providing conservative estimates of flux thimble tube wear and the interval between inspections is established such that no flux thimble tube is predicted to incur wear that exceeds the established acceptance criteria before the next inspection.

SRP-LR Section A.1.2.3.10 states that the operating experience of AMPs that are existing programs, including past corrective actions resulting in program enhancements or additional programs, should be considered and past failure would not necessarily invalidate an AMP because the feedback from operating experience should have resulted in appropriate program enhancements or new programs. It also states that this information should provide objective evidence to support the conclusion that the effects of aging will be managed adequately so that the structure and component-intended function(s) will be maintained during the period of extended operation.

Issue:

The LRA does not clearly indicate the root cause(s) that resulted in the repositioning and replacement of the 25 flux thimble tubes. In addition, the LRA does not provide the inspection and evaluation results associated with the corrective actions in order to demonstrate to the staff the adequacy of the program-defined wear projection methodology, inspection frequency, and acceptance criteria for the tube wall loss.

Request:

1. Describe the root cause(s) that led to the repositioning and replacement of the 25 flux thimble tubes. As part of the response, clarify whether any aging effect, other than loss of material due to wear, caused the repositioning and replacement of the thimble tubes.

2. Identify how many flux thimble tubes of Unit 2 were repositioned and/or replaced during each refueling outage. Specifically, describe whether or not any of the thimble tubes required repositioning more than once with or without subsequent replacement. If so, how was this factored back into the trending basis.

3. Compare Unit 1 and Unit 2 in terms of the extent and severity of the flux thimble tube wear. If Units 1 and 2 do not indicate comparable extent or severity of flux thimble tube wear, describe the applicant's engineering evaluation to identify the cause of the difference between the units, including identification of any corrective actions that have been taken, in view of the engineering evaluation.

4. Based on the evaluation and information addressed above, demonstrate that the applicant's program has adequately implemented the information and lessons obtained from the operating experience. If the foregoing evaluation of the operating experience identifies an item to be further implemented as a program enhancement, describe the item and applicant's enhancement associated with it.

STPNOC Response

1. Flux thimble tubes were repositioned and replaced due to wall thinning. The wall thinning is a result of flow-induced vibration. The flux thimble tube wear occurs generally at locations associated with geometric discontinuities or area changes along the flow path, such as areas near the lower core plate, the core support forging, the lower tie plate, the upper tie plate, and the vessel penetration. No other aging effect has been observed.

2. The following table summarizes the history of flux thimble tube repositioning and replacement in Unit 2:

Outage	Action
2RE4	12 repositioned
2RE5	3 capped/1 repositioned
2RE6	3 replaced/1 repositioned
2RE7	4 repositioned
2RE8	4 replaced
2RE9	5 repositioned
2RE10	8 replaced/8 repositioned
2RE11	1 repositioned
2RE12	2 repositioned
2RE13	10 replaced
2RE14	None

The flux thimble tubes in the table above were repositioned approximately 2½ inches because of wear or for fitting replacements. Some flux thimble tubes were repositioned more than once due to wear, but these have all been replaced. In addition to the above, some flux thimble tubes have been repositioned approximately ½ inch for fitting replacements. All replacement flux thimble tubes are chrome plated and have shown no wear.

The two most recent wear measurements for each flux thimble tube are trended to obtain a projected wear and compare with the result of the next wear measurement of the flux thimble. If the projected wear is greater than the acceptance criteria, then the flux thimble is replaced, repositioned sufficiently to prevent further wear in the same location, or removed from service by capping it.

3. The history of flux thimble tube wear for Units 1 and 2 has been considerably different. The original flux thimble tubes in Unit 1 had an outer diameter of 0.313 inch. These thimble tubes were replaced in 1989 with flux thimble tubes having an outer diameter of 0.385 inch and a larger wall thickness. In addition, sleeves were installed in the instrument column. Since that time, no actions have been required for flux thimble tube wear in Unit 1.

   The original flux thimble tubes in Unit 2 had the larger outer diameter of 0.385 inch and the instrument column was machined differently than Unit 1 so that sleeves were not installed in Unit 2. The wear of the flux thimble tubes in Unit 2 has required multiple flux thimble tubes to be repositioned, replaced, or removed from service.

   In WCAP-12866, Bottom Mounted Instrumentation Flux Thimble Wear, Westinghouse noted that there were several cases where the flux thimble tube wear was significantly different between plants which were essentially identical other than that some of the plants were equipped with instrumentation column sleeves. The plants equipped with the sleeves exhibited significantly less flux thimble tube wear than those plants that were not so equipped. The sleeves had been installed to make the inside diameter of the instrumentation columns the same as for the un-sleeved plants. Therefore, sleeves could not be installed in the un-sleeved plants. It is believed that this is the cause of the different flux thimble wear between the two STP Units and that no additional corrective action is practical for this condition.

4. Operating experience has been incorporated to reduce the flux thimble tube wear. In Unit 1, all the flux thimble tubes were replaced in 1989 with flux thimble tubes having a larger outer diameter. Since that time, the wear in the Unit 1 flux thimble tubes has been minor. In Unit 2, the flux thimble tubes exhibited wear even though they had the larger diameter. As it has become necessary to replace flux thimble tubes in Unit 2, they have been replaced with chrome plated flux thimble tubes. Twenty-five flux thimble tubes have been replaced to date, and have shown essentially no wear such that further re-positioning or replacement has not been required.

RAI B3.1-1

Background:

The "corrective actions" program element of GALL AMP X.M1, "Fatigue Monitoring," states that acceptable corrective actions include repair of the component replacement of the component, and a more rigorous analysis of the component to demonstrate that the design code limit will not be exceeded during the period of extended operation.

Issue:

LRA Section B3.1, "Metal Fatigue of Reactor Coolant Pressure Boundary," proposes an enhancement to the "corrective actions" program element, which states that if the CUF has approached 1.0, then further actions for cumulative fatigue usage actions limits may be invoked. As part of the enhancement, the applicant included seven options as acceptable corrective actions for the program to take when the CUF has approached 1.0. Four of the proposed corrective actions are beyond the recommendations in GALL AMP X.M1, as follows:

1) Determine whether the scope of the management program must be enlarged
2) Enhance fatigue managing
3) Modify plant operating practices
4) Perform a flaw tolerance evaluation and impose component-specific inspections

It is not clear to the staff if these four additional options for corrective actions to prevent CUF or CUF_{en} from exceeding the design code limit will be taken when the applicant's action limit is reached or when the fatigue usage has approached 1.0.

The staff noted that LRA Section A2.1, which provides the UFSAR Supplement for the Metal Fatigue of Reactor Coolant Pressure Boundary Program, does not describe this proposed enhancement. Commitment No. 30 in LRA Table A4-1 provides a summary statement for each enhancement. However, for the enhancement to the "corrective actions" program element, the applicant did not provide sufficient details in Commitment No. 30 that describe the corrective actions to be invoked if a component approaches a cycle counting action limit or a fatigue usage action limit.

Request:

- Clarify if the corrective actions, as described above, are applicable when CUF or CUF_{en} has approached 1) the applicant's action limits or 2) the Code design limit of 1.0. If these corrective actions are applicable to the latter, describe and justify how the use of these four options for corrective actions will prevent the CUF or CUF_{en} from exceeding the design code limit during the period of extended operation. If appropriate, revise the LRA accordingly.

- Provide clarifications for Commitment No. 30 to describe the corrective actions to be invoked if a component approaches a cycle counting action limit, a fatigue usage action limit, and when CUF or CUF_{en} has approached 1.0. Or justify that the UFSAR supplement in LRA Section A2.1 provides a sufficiently comprehensive summary description of the Metal Fatigue of Reactor Coolant Pressure Boundary Program that prevents the usage factor from exceeding the design code limit during the period of extended operation.

STPNOC Response

Corrective actions are initiated when an action limit is reached. Action limits are established to ensure that correction actions are completed prior to exceeding the Code CUF design limit of 1.0.

LRA Appendix B3.1, Table A4-1 Commitment No. 30 and LRA Basis Document AMP X.M1 (B3.1) Metal Fatigue of Reactor Coolant Pressure Boundary, will be revised to clarify the corrective actions to be invoked if a component cycle counting action limit is reached and the corrective actions to be invoked if a CUF or CUF_{en} action limit is reached. These corrective actions will include repair of the component, replacement of the component or a more rigorous analysis for the component to demonstrate that the design code limit will not be exceeded during the period of extended operation

RAI B3.1-2

Background:

SRP-LR Section A 1.2.3.10 states that if the aging management program is an existing program, operating experience of the program should provide objective evidence to support the conclusion that the effects of aging will be managed adequately so that the intended function(s) will be maintained during the period of extended operation.

Issue:

LRA Section B3.1, "Metal Fatigue of Reactor Coolant Pressure Boundary," discusses the operating experience associated with fatigue issues that are focused primarily on industry initiatives and NRC/vendor information that caused the applicant to assess thermal stratification of the pressurizer surge line and thermal fatigue cracking in normal-isolated piping. During its audit, the staff reviewed the applicant's operating experience and condition reports, and noted that fatigue issues related to cycle-counting had occurred, such as when certain transient cycle counts (Loss of Charging with prompt restoration without loss of letdown flow and Cold Over-pressurization Mitigation Systems actuation) approached their respective action limits. The staff noted that LRA Section B3.1 did not discuss these in-service fatigue issues and the actions taken by the applicant.

Request:

Justify that objective evidence such as that referenced above, with examples and sufficient details from plant-specific experience, has been included in the "operating experience" program element of the Metal Fatigue of Reactor Coolant Pressure Boundary Program to support the conclusion that the effects of aging will be managed adequately during the period of extended operation

STPNOC Response

The STP Corrective Action Program database documents 11 occurrences of Loss of Charging (also known as Charging Flow Shutoff with Prompt Return-to-Service). The baseline count currently referenced in the STP LRA accounts for all but one of these occurrences. After a review of the plant instrument data, it is concluded that a Loss of Charging did not occur for this eleventh occurrence. The perturbations of charging flow are more characteristic of the Charging Flow Step Decrease and Return to Normal transient that assumes 24,000 occurrences for the design number of cycles. The cycle counting to date for this type of occurrence gives assurance that the perturbations of charging flow expected for 60 years are expected to be far less that the number of occurrences assumed in the fatigue analysis.

The STP Corrective Action Program database documents three occurrences of Cold Over-pressurization Mitigation Systems (COMS) activation. This is consistent with the baseline count currently referenced in the STP LRA.

Both of the events resulted in condition reports being initiated because STP exceeded a 30 percent alert limit. The purpose of that alert limit is to ensure that the transients accumulate at a rate less than that assumed in the design basis. The fatigue monitoring implemented for license renewal will incorporate cycle projections into the programs acceptance criteria. These projections will be based on accumulation history since the start of plant life and use long-term weighting (LTW) and short-term weighting (STW) factors to obtain the most accurate projections of future event behavior. Acceptance criteria will be 80 percent of the design number of transients. The conditions noted are not applicable to the license renewal Metal Fatigue of Reactor Coolant Pressure Boundary Program and are not cited in the LRA.

RAI B3.1-3

Background:

The staff noted that the applicant's Metal Fatigue of Reactor Coolant Pressure Boundary Program is based on GALL AMP X.M1, "Fatigue Monitoring," which is limited to the use of cycle-counting for CUF analyses (e.g. ASME Code, Section III CUF analyses and environmentally-assisted fatigue CUF analyses). The use of cycle counting to manage crack growth of either a postulated or existing macro flaw is not covered by GALL AMP X.M1.

Issue:

LRA Section 4.3.2.11 credits the Metal Fatigue of Reactor Coolant Pressure Boundary Program to manage the aging effects associated with the leak-before-break (LBB) TLAA, which was dispositioned in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 54.21(c)(1)(iii), that the effects of aging on the intended functions will be adequately managed for the period of extended operation. The applicant expanded the use of cycle counting for the LBB TLAA, which is a non-CUF type analysis, without including enhancements to the applicable program elements of this aging management program.

In addition, it is not clear to the staff if the applicant's basis for cycle counting design transients has been captured in the applicable documents (e.g. Technical Specifications, UFSAR, and cycle counting procedure) and describes the management of crack growth during the period of extended operation.

Request:

- Justify the use of cycle counting in the Metal Fatigue of Reactor Coolant Pressure Boundary Program for the LBB TLAA and its disposition in accordance with 10 CFR 54.21 (c)(1)(iii) without: 1) an update to the applicable documents (e.g. Technical Specifications, UFSAR, and cycle counting procedure) and 2) the inclusion of enhancements to the applicable program elements in the Metal Fatigue of Reactor Coolant Pressure Boundary Program.

- If enhancements and associated commitments are necessary, provide the following: 1) justification for the use of cycle counting activities, 2) clarification of the transients that require monitoring for this TLAA, 3) action limits associated with the assumed design transients, 4) corrective action(s) that will be taken if an action limit is reached, and 5) appropriate revisions to the LRA.

STPNOC Response:

The UFSAR will be updated in accordance with 10 CFR 54.29 to identify those transients used in the LBB analysis. LRA Appendix B3.1 and LRA Basis Document AMP X.M1 (B3.1), Metal Fatigue of Reactor Coolant Pressure Boundary program, Element 1 will be revised to identify the increase in the scope of the program to ensure the fatigue crack growth analyses, which support the leak-before-break (LBB) analyses, remain valid by counting the transients used in the analyses.

The following support the enhancements and associated commitments:

1) The cycle counting activity of the Metal Fatigue of Reactor Coolant Pressure Boundary Program is appropriate for management of the LBB analyses because the transients used are consistent with those used in the fatigue design basis.

2) The table below identifies the transients used in LBB analyses. These values were used to determine the program limiting values presented in LRA Table 4.3-2 with the exception of two transients that are not listed in LRA Table 4.3-2. The exceptions are the "Accumulator Actuation, Accident Operation" and "Reduce Temperature Return to Power" transients. The "Accumulator Actuation, Accident Operation" transient is a combination of the "Inadvertent RCS Depressurization" transient, which is monitored, and "LOCA," which is a faulted event; therefore the "Accumulator Actuation, Accident Operation" transient is being managed for the period of extended operation. The "Reduce Temperature Return to Power" transient was included in pressurizer surge line fatigue crack growth analysis. This transient was not incorporated into the design basis of STP (i.e. the transient is not included in the fatigue analyses of the Class 1 components or the UFSAR). This transient is designed to improve capabilities of the plant during load follow operations. STP does not practice load follow operations. Therefore, this transient, while included in the LBB analysis, is not applicable to STP operation.

Transients Used in the LBB Fatigue Crack Growth Analyses				
Transient	**LRA Table 4.3-2**	**Reactor Coolant Loop**	**Pressurizer Surge Line**	**Accumulator Lines**
Normal				
Heatup	200	200	200	-
Cooldown	200	200	200	-
Unit Loading at 5 percent of Full Power/min	3,000	13,200	13,200	13,200
Unit Unloading at 5 percent of Full Power/min	3,000	13,200	13,200	13,200
Reduce Temperature Return to Power	-	-	2,000	-
Step Load Increase of 10 percent of Full Power	2,000	2,000	2,000	2,000
Step Load Decrease of 10 percent of Full Power	2,000	2,000	2,000	2,000
Large Step Load Decrease with Steam Dump	200	200	200	-
Steady State Fluctuations, Initial	1.5×10^5	1.5×10^5	3.15×10^6 (Note)	3.2×10^6 (Note)
Steady State Fluctuations, Random	3.0×10^6	3.0×10^6		
Feedwater Cycle at Hot Shutdown	2,000	2,000	2,000	2,000
Loop Out of Service, Normal (Active) Loop Shutdown	80	80	80	-
Loop Out of Service, Normal (Inactive) Loop Startup	70	70	70	-
Unit Loading Between 0-15 percent of Full Power	500	500	500	-
Unit Unloading Between 0-15 percent of Full Power	500	500	500	-
Boron Concentration Equalization	26,400	26,400	26,400	-
Refueling	80	80	80	80
Primary Side Leak Test	U1-120 U2-200	200	200	-
Secondary Side Leak Test	80	80	80	-
Tube Leak Test	Type I: 400 Type II: 200 Type III: 120 Type IV: 80	800	800	-
Turbine Roll Test	20	20	20	20
Upset				
Loss of Load (Without Immediate Reactor Trip)	80	80	80	-
Loss of Power (Blackout; Loss of Offsite AC Power with Natural Circulation in the RCS)	40	40	40	-
Partial Loss of RCS Flow (Loss of One RCP)	80	80	80	-
Reactor Trip from Full Power, without Cooldown.	230	230	230	-
Reactor Trip from Full Power, with Cooldown, without Safety Injection	160	160	160	160
Reactor Trip from Full Power, with Cooldown, with Safety Injection	10	10	10	-
Inadvertent RCS Depressurization	20	20	20	20
Inadvertent RCS Depressurization due	10	10	10	-

Transients Used in the LBB Fatigue Crack Growth Analyses				
Transient	**LRA Table 4.3-2**	**Reactor Coolant Loop**	**Pressurizer Surge Line**	**Accumulator Lines**
to Inadvertent Auxiliary Spray				
Inadvertent Startup of an Inactive RCS Loop	10	10	10	-
Control Rod Drop	80	80	80	80
Inadvertent ECCS Actuation (No Safety Injection)	60	60	60	-
Operating Basis Earthquake (OBE)	5	5	5	-
Excessive Feedwater Flow	30	30	30	-
Test				
Primary Side Hydrostatic Test	1	10	-	-
Secondary Side Hydrostatic Test (each generator)	10	10	-	-
Auxiliary Conditions				
Inadvertent Accumulator Blowdown	4	-	-	4
RHR Operation	200	-	-	200
High Head Safety Injection	54	-	-	110
Accumulator Actuation, Accident Operation	-	-	-	21

Note: For normal steady state fluctuations, there is no differentiation between initial and random transients for the pressurizer surge line and the accumulator lines.

3) The action limits are set at 80 percent of the design value, consistent with the action limits associated with the management of fatigue usage.

4) Corrective actions include a review the fatigue crack growth analyses that support the LBB exemptions to ensure that the analytical bases remain valid. Re-analysis of a fatigue crack growth analysis must be consistent with, or reconciled to, the originally submitted analysis and receive the same level of regulatory review as the original analysis. LRA Appendix B3.1 and LRA Basis Document AMP X.M1 (B3.1), Metal Fatigue of Reactor Coolant Pressure Boundary program will be revised to incorporated these corrective actions into the enhancement associated with Element 7, Corrective Actions.

5) LRA Appendix B3.1 and LRA Basis Document AMP X.M1 (B3.1), Metal Fatigue of Reactor Coolant Pressure Boundary program will also be revised to reflect the enhancements to Element 1, Scope of Program, and Element 7, Corrective Actions, described in items 1 and 4 above.

RAI B3.1-4

Background:

GALL AMP X.M1, "Fatigue Monitoring," states that corrective actions are provided to prevent the usage factor from exceeding the design code limit during the period of extended operation. LRA Section B3.1, "Metal Fatigue of Reactor Coolant Pressure Boundary," proposes an enhancement to the "corrective actions" program element, which includes several corrective actions to be invoked when a cycle counting action limit or a CUF action limit is reached.

Issue:

The enhancement to the "corrective actions" program element states that the counting action limits are based on a somewhat-arbitrary cycle count that does not accurately indicate approach to the CUF =1.0 fatigue limit. It is not clear to the staff what the "somewhat-arbitrary cycle count" in the applicant's program references and how it impacts the effectiveness of the program to ensure the design limit on fatigue usage will not be exceeded.

In addition, this enhancement states that one acceptable corrective action if a CUF action limit is reached is to enhance fatigue managing to confirm continued conformance to the code limit. It is not clear to the staff how the applicant will "enhance fatigue managing" and whether this action will prevent the cumulative usage factor from exceeding the design limit during the period of extended operation.

Request:

- Identify what is the "somewhat-arbitrary cycle count" in the Metal Fatigue of Reactor Coolant Pressure Boundary Program and justify that the "somewhat-arbitrary cycle count" will not impact the program's capability to prevent the usage factor from exceeding the design code limit during the period of extended operation.
- Identify the proposed actions to "enhance fatigue managing" and justify that they will be effective to prevent the usage factor from exceeding the design code limit during the period of extended operation.

STPNOC Response:

The statement of a "somewhat-arbitrary cycle count" is in reference to the fact that the fatigue analyses are based on the number of design transients specified in UFSAR Table 3.9-8. These are not values that result in a CUF equal to 1.0; therefore, when a design number of a transient is reached, there is inherent margin for measures to be taken to prevent the usage factor from exceeding the Code limit of 1.0. . LRA Basis Document X.M1 (B3.1) Corrective Action (Element 7) Enhancement will be revised to read:

> ~~Since the counting action limits are based on a somewhat-arbitrary cycle count that does not accurately indicate approach to the CUF = 1.0 fatigue limit,~~ These preliminary actions are designed to determine how close the approach is to the 1.0 limit, and from those determinations, set new action limits. If the CUF has approached 1.0 then further actions described below for cumulative fatigue usage action limits may be invoked.

The proposed action to "enhance fatigue managing" will be deleted as part of the changes to the LRA described in the response to RAI B3.1-1. For LRA Appendix B3.1, the Section on Enhancements for *Corrective Actions (Element 7)* regarding when the CUF action limit is reached will be revised as follows:

> If a CUF action limit is reached acceptable corrective actions include:
>
> ~~1) Determine whether the scope of the management program must be enlarged to include additional affected reactor coolant pressure boundary locations. This determination will ensure that other locations do not approach design limits without an appropriate action.~~
>
> ~~2) Enhance fatigue managing to confirm continued conformance to the code limit.~~

1~~3~~) Repair the component.

2~~4~~) Replace the component. If a limiting component is replaced, assess the effect on locations monitored by the program. If a limiting component is replaced, resetting its cumulative fatigue usage factor to zero, a component which was previously bounded by the replaced component will become the limiting component and may need to be monitored.

3~~5~~) Perform a more rigorous analysis of the component to demonstrate that the design code limit will not be exceeded during the period of extended operation.

~~6) Modify plant operating practices to reduce the fatigue usage accumulation rate.~~

~~7) Perform a flaw tolerance evaluation and impose component specific inspections, under ASME Section XI Appendices A or C (or their successors), and obtain required approvals by the NRC.~~

RAI B3.1-5

Background:

The "scope of program" program element of GALL AMP X.M1, "Fatigue Monitoring," recommends that the program should include, for a set of sample reactor coolant system components, fatigue usage calculations that consider the effects of the reactor water environment. This sample set should include the locations identified in NUREG/CR-6260 and additional plant-specific component locations in the reactor coolant pressure boundary if they may be more limiting than those considered in NUREG/CR-6260.

Issue:

During its audit and review of LRA Section B3.1, "Metal Fatigue of Reactor Coolant Pressure Boundary" and supporting program basis documents, the staff did not find any identification of additional component locations other than those from NUREG/CR-6260, or an evaluation that confirmed the NUREG/CR-6260 locations were bounding for the applicant's site. Furthermore, the staff noted that the applicant's plant-specific configuration may contain locations that should be analyzed for the effects of the reactor coolant environment other than those identified in NUREG/CR-6260.

Request:

- Justify that the plant-specific locations listed in LRA Table 4.3-8 are bounding for the generic NUREG/CR-6260 components.

- Confirm and justify that the locations selected for environmentally assisted fatigue analyses in LRA Table 4.3-8 consists of the most limiting locations for the plant (beyond the generic components identified in the NUREG/CR-6260 guidance). If these locations are not bounding, clarify the locations that require an environmentally assisted fatigue analysis and the actions that will be taken as part of the Metal Fatigue of Reactor Coolant Pressure Boundary Program for these additional locations. If the identified limiting location consists of nickel alloy, state whether the methodology used to perform the environmentally-assisted fatigue calculation for nickel alloy is consistent with NUREG/CR-6909. If not, justify the method chosen.

STPNOC Response:

No additional reactor coolant pressure boundary components were considered for inclusion in the environmentally-assisted fatigue analyses beyond those assessed in LRA Table 4.3-8.

A new commitment (see Enclosure 2 to this letter) will be added to LRA Table A4.-1 which states:

> Prior to the period of extended operation STP will perform a review of design basis ASME Class 1 component fatigue evaluations to determine whether the NUREG/CR-6260-based components that have been evaluated for the effects of the reactor coolant environment on fatigue usage are the limiting components for the STP configuration. If more limiting components are identified, the most limiting component will be evaluated for the effects of the reactor coolant environment on fatigue usage. If the limiting location consists of nickel alloy, the methodology for nickel alloy in NUREG/CR-6909 will be used to perform the environmentally-assisted fatigue calculation. The additional evaluation will be performed through the Metal Fatigue of Reactor Coolant Pressure Boundary Program in accordance with 10 CFR 54.21(c)(1)(iii).

Enclosure 2

New Regulatory Commitment

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.

Table A4-1 License Renewal Commitments

Item #	Commitment	LRA Section	Implementation Schedule
34	Prior to the period of extended operation STP will perform a review of design basis ASME Class 1 component fatigue evaluations to determine whether the NUREG/CR-6260-based components that have been evaluated for the effects of the reactor coolant environment on fatigue usage are the limiting components for the STP configuration. If more limiting components are identified, the most limiting component will be evaluated for the effects of the reactor coolant environment on fatigue usage. If the limiting location consists of nickel alloy, the methodology for nickel alloy in NUREG/CR-6909 will be used to perform the environmentally-assisted fatigue calculation. The additional evaluation will be performed through the Metal Fatigue of Reactor Coolant Pressure Boundary Program in accordance with 10 CFR 54.21 (c)(1)(iii).	B3.1	Prior to the period of extended operation