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

August 15, 2011

Mr. G.T. Powell, Vice President
Technical Support and Oversight
STP Nuclear Operating Company
P.O. Box 289
Wadsworth, TX 77483

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

Dear Mr. Powell:

By letter dated October 25, 2010, STP Nuclear Operating Company, submitted an application pursuant to Title 10 of the *Code of Federal Regulations* Part 54 for review by the U.S. Nuclear Regulatory Commission (NRC or the staff), to renew operating licenses NPF-76 and NPF-80 for South Texas Project, Units 1 and 2. The staff is reviewing the information contained in the license renewal application and has identified, in the enclosure, areas where additional information is needed to complete the review.

These requests for additional information were discussed with Arden Aldridge, and a mutually agreeable date for the response is within 30 days from the date of this letter. If you have any questions, please contact me at 301-415-3873 or by e-mail at john.daily@nrc.gov.

Sincerely,

John W. Daily

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

Docket Nos. 50-498 and 50-499

Enclosure:
As stated

cc w/encl: Listserv

SOUTH TEXAS PROJECT, UNITS 1 AND 2
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.

ENCLOSURE

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.

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% 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. B6.50 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.

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.

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.

Justify the adequacy of the sampling methodology for butt welds if the inspection sample sizes are less than ten percent (up to 25 welds) per unit, which is the recommendation in GALL AMP XI.M35. Revise the LRA, including UFSAR supplement, consistent with the RAI response.

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.

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.

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.

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.

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.

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.

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.

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.

August 15, 2011

Mr. G.T. Powell, Vice President
Technical Support and Oversight
STP Nuclear Operating Company
P.O. Box 289
Wadsworth, TX 77483

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

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These requests for additional information were discussed with Arden Aldridge, and a mutually agreeable date for the response is within 30 days from the date of this letter. If you have any questions, please contact me at 301-415-3873 or by e-mail at john.daily@nrc.gov.

Sincerely,

/RA/

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

Docket Nos. 50-498 and 50-499

Enclosure:
As stated

cc w/encl: Listserv

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ADAMS Accession No.: ML11214A027 *concurrence via e-mail

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DATE	08/04/2011	08/8/2011	08/8/2011	08/8/2011	08/15/2011

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Letter to G.T. Powell from John W. Daily dated August 15, 2011

SUBJECT: 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 (TAC NOS. ME4936 AND ME4937)

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