SeabrookNPEm Resource

From: Plasse, Richard
Sent: Monday, November 15, 2010 1:51 PM
To: Cliche, Richard
Subject: FW: Transmittal of RAIs relating to Seabrook (AMP Audit)
Attachments: SBK AMP RAIs Reactor Head Closure Studs (TRP 3)- Pan SMin OYeedozier - clean.docx; SBK AMP RAI Small Bore (TRP 36) - Mintz - Fu - Oyeedozier - clean.docx; SBK AMP RAIs FatigueMonitoring (TRP 55) Chopra Ng - OYee-dozier.docx; SBK AMP RAIs ISI (TRP 1) - Mintz BFu OYee docxdozier - clean.docx

Importance: High

Draft RAIs

Hearing Identifier: Seabrook_License_Renewal_NonPublic
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Mail Envelope Properties (Richard.Plasse@nrc.gov20101115135000)

Subject: FW: Transmittal of RAIs relating to Seabrook (AMP Audit)
Sent Date: 11/15/2010 1:50:58 PM
Received Date: 11/15/2010 1:50:00 PM
From: Plasse, Richard

Created By: Richard.Plasse@nrc.gov

Recipients:
"Cliche, Richard" <Richard.Cliche@fpl.com>
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Files	Size	Date & Time
MESSAGE	12	11/15/2010 1:50:00 PM
SBK AMP RAIs Reactor Head Closure Studs (TRP 3)- Pan SMin OYeedozier - clean.docx	24559	
SBK AMP RAI Small Bore (TRP 36) - Mintz - Fu - Oyeedozier - clean.docx	20539	
SBK AMP RAIs FatigueMonitoring (TRP 55) Chopra Ng - OYee-dozier.docx	32587	
SBK AMP RAIs ISI (TRP 1) - Mintz BFu OYee docxdozier - clean.docx	16909	

Options
Priority: High
Return Notification: No
Reply Requested: No
Sensitivity: Normal
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Seabrook Reactor Head Closure Studs Program

RAI B.2.1.3-1 (#193)

Background
The "preventive actions" program element of GALL AMP XI.M3, "Reactor Head Closure Studs," references the guidance outlined in RG 1.65 originally issued in 1973. RG 1.65, Rev. 1 was issued in April, 2010 and includes using bolting material for closure studs that has a measured yield strength less than 150 ksi, which is resistant to stress corrosion cracking.

LRA Section B.2.1.3 states that the Seabrook reactor head closure studs are manufactured from SA-540, Class 3, Grade B24 material and the maximum tensile strength of the material is less than 170 ksi as recommended in GALL Report, Rev. 1.

Issue
LRA Section B.2.1.3 does not include the preventive action using stud materials with a measured yield strength level less than 150 ksi in comparison with RG 1.65, Rev. 1. The staff needs to confirm whether the applicant's program considers the strength levels of reactor head closure stud materials as addressed in the RG 1.65, Rev. 1 to adequately manage stress corrosion cracking.

Request
1) Clarify whether the measured yield strength of the reactor head closure stud material used at Seabrook Station exceeds 150 ksi.

2) Are there program provisions that would preclude use of materials with yield strength greater than 150 ksi? If not, or if the reactor head closure stud material has a yield strength level greater than or equal to 150 ksi, justify the adequacy of the Reactor Head Closure Studs Program to manage stress corrosion cracking in the high-strength material.

RAI B.2.1.3-2 (#194)

Background
The program description of GALL AMP XI.M3, "Reactor Head Closure Studs," states that the recommended program includes inservice inspection to detect cracking, loss of material and coolant leakage from reactor head closure studs. The "preventive actions" program element of GALL AMP XI.M3 also includes using manganese phosphate or other acceptable surface treatments and stable lubricants. LRA Section B.2.1.3 indicates that a station approved lubricant is utilized during installation/removal of the studs that does not contain molybdenum disulfide (MoS_2).

Issue
Operating Experience No. 2 described in LRA Section B.2.1.3 states that discoloration was reported on some of the reactor head closure studs during Refueling Outage 8 in 2002, and that the discoloration was due to the lubricant used for stud removal and was considered not an indication of stud degradation. During the staff's audit, the applicant also stated that the substance applied on the studs was WD-40. The staff needs to confirm whether the discoloration is related to an age-related degradation. The staff also

needs to clarify whether WD-40 is a stable lubricant at the operating temperatures and compatible with reactor bolting materials and environment.

Request

1) Clarify what the root cause for the discoloration on the studs was and whether the discoloration has been repeatedly observed. In its response, provide further justification why the observed discoloration is not associated with an aging effect that requires management during the period of extended operation, such as loss of material due to corrosion or wear. If the discoloration is associated with an aging effect, justify how it will be managed during the period of extended operation.

2) Provide the service temperature range of the lubricant based on its technical specification or equivalent. In addition, compare the service temperature with the operating temperatures of the reactor head closure studs. In view of the foregoing evaluation, further clarify whether the lubricant is stable at the operating temperatures and is compatible with the stud and vessel materials and with the surrounding environment.

RAI B.2.1.23-1

Background

The "monitoring and trending" program element of GALL AMP XI.M35, "One-Time Inspection of ASME Code Class 1 Small-Bore Piping" states that a one-time volumetric inspection is an acceptable method for confirming the absence of cracking of ASME Code Class 1 small-bore piping. The GALL Report also states that the inspection of small bore piping should be performed at a sufficient number of locations to assure an adequate sample and that this number, or sample size, will be based on susceptibility, inspectability, dose considerations, operating experience, and limiting locations of the total population of ASME Code Class 1 small-bore piping locations. GALL AMP XI.M35 states that MRP-146 provides guidelines for identifying piping susceptible to one subset of cracking, including thermal stratification or turbulent penetrations. The applicant's program states that it will inspect for cracking in ASME Code Class 1 small-bore piping using qualified volumetric examination techniques, if available, and that if the non-destructive volumetric examination techniques have not been qualified, Seabrook Station will have the option to remove the weld for destructive examination. The applicant stated during the staff's audit that it will inspect 10% of the butt welds and 10% of the socket welds. In addition, the applicant stated that it may not inspect certain welds based on inaccessibility or high radiation exposure.

Issue

It is not clear to the staff if the applicant will either conduct an acceptable volumetric inspection or plan to do destructive examination. Based on the language in the applicant's program basis document, the staff noted that if an acceptable volumetric exam is not available before the period of extended operation, the applicant will have the option to perform destructive exams. In addition, the staff noted that the applicant proposed to inspect weld locations that are susceptible to SCC and cyclical loading, but the sampling methodology for the inspection was not presented. It was also not clear to the staff what part of the socket welds the applicant plans to inspect.

Request

1. Clarify and justify the use of destructive examination as an "option" within the program and FSAR supplement if an "acceptable" volumetric method isn't available.
2. Clarify what is meant by an "acceptable" volumetric inspection and justify the use of a volumetric technique if it is not consistent with GALL AMP XI.M35 recommendations.
3. Describe the methodology for choosing the types of welds to inspect and how this methodology will ensure the AMP adequately manages the effects of relevant forms of cracking during the period of extended operation.
4. Provide clarification on the methodology that will be used to manage inaccessible or high radiation exposure welds within the scope of the One-Time Inspection of ASME Code Class 1 Small-Bore Piping Program and justify this methodology.
5. Clarify the proposed examination volume approach for socket welds, and justify that the examination volume is sufficient and capable of detecting cracking in the subject socket welds.

Seabrook Station LRA Metal Fatigue AMP RAI

RAI B.2.3.1-1

Background

The scope of the Metal Fatigue of Reactor Coolant Pressure Boundary Program includes both nuclear steam supply system (NSSS) and non-NSSS components and transients in UFSAR Section 3.9.1.1 that are required to be tracked. LRA Section 4.3 states that the metal fatigue TLAAs that are evaluated in the LRA fall into the following three categories:

category (a) - Explicit fatigue analyses for NSSS pressure vessels and components prepared in accordance with ASME Section III, Class A or Class 1 rules developed as part of the original design.

category (b) - Supplemental explicit fatigue analyses for piping and components that were prepared in accordance with ASME Section III rules to evaluate transients that were identified after the original design analyses were completed, such as pressurizer surge line thermal stratification, and also include reactor vessel internal component fatigue analyses.

category (c) - New fatigue analyses (also in accordance with ASME Section III, Class 1 rules) prepared for license renewal to evaluate the effects of the reactor water environment on the sample of high fatigue locations applicable to newer vintage Westinghouse Plants, as identified in Section 5.5 of NUREG/CR-6260, and using the methodology presented in LRA Section 4.3.4.

In addition, LRA Section 4.3.1 states that the most limiting numbers of transients used in these NSSS component analyses are shown in Table 4.3.1-2, and are considered to be design limits. The staff confirmed that these transients are consistent with those listed in UFSAR Table 3.9(N)-1.

Issue

LRA Table 4.3.1-2 lists more plant design transients than those identified in Technical Specification (TS) 5.7 and TS Table 5.7-1. For example, in the TS table, normal condition transients include only plant heatup and shutdown; upset set transients include only loss of load w/o turbine roll, loss of all offsite power, partial loss of flow, and reactor trip from full power; faulted transients include large steam line break; and test transients include primary and secondary side hydrostatic test, and primary side leak test. It is not clear to the staff whether the design CUF fatigue analyses for NSSS pressure vessels and components were based on the design transients listed in TS Table 5.7-1 or the non-TS transients that were included in LRA Table 4.3.1-2 and UFSAR Table 3.9(N)-1.

In the event that a transient that is listed in LRA Table 4.3.1-2 but not in the TS occurs, it is not clear to the staff how the transient will be accounted for in accordance with the Metal Fatigue of Reactor Coolant Pressure Boundary Program during the period of extended operation.

The “parameters monitored/inspected” program element of GALL AMP X.M1 states the program monitors all plant transients that cause cyclic strains, which are significant contributors to the fatigue usage factor.

Request
(1) Clarify whether the category (a) fatigue analysis and the category (b) supplemental fatigue analysis were based on transients from TS Table 5.7-1 or LRA Table 4.3.1-2 [and in UFSAR Table 3.9(N)-1].

(2) Confirm that the plant-specific cycle counting procedure ensures those design transients that are listed in LRA Table 4.3.1-2 but not in TS 5.7 will be tracked and monitored (i.e., counted) in accordance with the Metal Fatigue of Reactor Coolant Pressure Boundary Program, during the period of extended operation. If these transients are not monitored during the period of extended operation, justify why they are not monitored by the Metal Fatigue of Reactor Coolant Pressure Boundary Program, consistent with the “parameters monitored/inspected” program element.

RAI B.2.3.1-2

Background
The scope of the Metal Fatigue of Reactor Coolant Pressure Boundary Program includes both nuclear steam supply system (NSSS) and non-NSSS components and transients in UFSAR Section 3.9.1.1 that are required to be tracked. The applicant stated that the most limiting numbers of transients used in these NSSS component analyses are shown in LRA Table 4.3.1-2, and are considered to be design limits.

Issue
The staff noted that the transients are termed differently in the LRA, UFSAR, and relevant documents that were reviewed during the staff's audit. For example, upset condition transients such as "inadvertent startup of an inactive loop" or "inadvertent emergency core cooling system actuation" are referred to differently in these documents. The staff also noted, during its audit, that the applicant's program basis document includes auxiliary transients such as "charging and letdown flow shutoff and return" or "letdown flow step decrease and return", however, these transients are not included in the list of design transients provided in LRA Table 4.3.1-2.

Request
(a) Justify that the difference of designations for the transients between LRA Table 4.3.1-2 and CUF analyses in the applicant's program basis document should not be aligned. Clarify and justify how the Metal Fatigue of Reactor Coolant Pressure Boundary Program and the associated on-site procedure will be capable of tracking transient occurrences to ensure that the design limit of 1.0 is not exceeded and that any assumptions that are made in the fatigue CUF analyses remain valid, if the designations for the transients are not consistent between the LRA, the UFSAR, and other relevant documents.

(b) Clarify the significance of the auxiliary transients used in fatigue CUF analyses, and explain how these transients are accounted for by the list of design transients provided in LRA Table 4.3.1-2.

RAI B.2.3.1-3

Background
The LRA Section B.2.3.1 and Commitment No. 41 state that the following enhancement will be made prior to entering the period of extended operation:

> The Metal Fatigue of Reactor Coolant Pressure Boundary Program will be enhanced to include additional transients beyond those defined in the TS and UFSAR.

Issue
The Metal Fatigue of Reactor Coolant Pressure Boundary Program does not identify these additional design transients that are monitored beyond those defined in the TS and UFSAR. The staff also noted that the applicant's program does not provide any description or the significance of these additional transients. The applicant's program also does not identify the components that these additional transients affect, specifically those CUF TLAAs in LRA Section 4.3 that the applicant dispositioned 10 CFR 54.21(c)(1)(iii)

Request
(a) Identify all additional design transients that are monitored by the Metal Fatigue of Reactor Coolant Pressure Boundary Program and justify why these additional transients need to be monitored. Discuss the significance of these additional transients to the TLAAs that have been identified in LRA Section 4.3.

(b) Clarify how these additional transients relate to the Technical Specification 5.7 and transients analyzed for in UFSAR Section 3.9.

(c) Clarify whether these transients were included in the new environmentally-assisted fatigue analysis evaluations that were prepared for license renewal in LRA Section 4.3.4. If they were not included, provide justification why these transients are significant only for those analyses in the CLB and not significant for the analyses performed for the period of extended operation.

RAI B.2.3.1-4

Background
In LRA Section B.2.3.1, the applicant stated that the Metal Fatigue of Reactor Coolant Pressure Boundary Program monitors and tracks the number of transient cycles to ensure that the CUF for select reactor coolant system components remains less than 1.0 through the period of extended operation. The applicant also stated the program ensured the environmental effect on fatigue sensitive locations are addressed. Locations with CUF approaching the design limit are reanalyzed, inspected, repaired, or replaced as necessary in accordance with applicable design codes. LRA Section B.2.3.1 states that pre-established action limits will permit completion of corrective actions before the design basis number of events is exceeded, and before the CUF, including environmental effects, exceeds the ASME Code limit of 1.0.

Issue
It is not clear to the staff if the Metal Fatigue of Reactor Coolant Pressure Boundary Program will perform cycle-counting, cycle-based fatigue monitoring, or stress-based fatigue monitoring for reactor coolant pressure boundary components (including the environmentally-assisted fatigue). The Metal Fatigue of Reactor Coolant Pressure Boundary Program does not provide details regarding the action limits that are set on design basis transient cycle counting activities or on CUF monitoring activities, or the corrective actions that will be implemented if an action limit of cycle counting or CUF monitoring is reached.

The staff has noted that LRA Section 4.3 sets a design limit of 1.0 for CUF analyses and environmentally-adjusted CUF analyses but the design limit for high energy line break locations is set to a value of 0.1. Furthermore, in order to maintain a design limit of 1.0, it should be noted that the action limit for cycle counting or CUF monitoring can be different if the same transient is used in a CUF analyse of a component or an environmentally-adjusted CUF analyses of another component.

Request

Define and justify the “action limit or limits” that will be used by the Metal Fatigue of Reactor Coolant Pressure Boundary Program for:

- design basis CUF values for Class 1 components and any non-Class 1 components evaluated to Class 1 component CUF requirements

- environmentally-assisted CUF for the program’s NUREG/CR-6260 equivalent or bounding locations, and

- Class 1 components that are within the scope of the applicant’s high-energy line break analyses for Class 1 components.

RAI B.2.3.1-5

Background

The “corrective actions” program element in GALL X.M1, “Metal Fatigue of Reactor Coolant Pressure Boundary” 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 extended period of operation.

Issue

In LRA Section B.2.3.1, the applicant stated that corrective actions may encompass one of several activities below:

1. Reanalyze affected component(s) for an increase in the number of that specific transient while accounting for other component-affecting plant transients that may be projected not to achieve their analyzed levels.

2. Perform a fracture mechanics evaluation of a postulated flaw in affected plant components, which, when coupled with an inservice inspection program, will serve to demonstrate flaw tolerant behavior.

3. Repair the affected component.

4. Replace the affected component.

Request

Provide a justification for the corrective action, to perform a fracture mechanics evaluation, which is not consistent with the recommendations of the “corrective actions” program element of GALL AMP X.M1

RAI B.2.3.1-6

Background

The “Detection of Aging Effects” program element in GALL X.M1 (“Metal Fatigue of Reactor Coolant Pressure Boundary”) states that the aging management program provides periodic update of the fatigue usage calculations. The LRA Section B.2.3.1 also states that the Metal Fatigue of Reactor Coolant Pressure Boundary Program will be enhanced to use a software program to count transients to monitor cumulative usage on selected components. The applicant also included this enhancement in LRA Commitment No. 42 in LRA Table A.3. LRA Section B.2.3.1 also states that “The program includes generation of a periodic fatigue monitoring report, including a listing of transient events, cycle summary event details, cumulative usage factors, a detailed fatigue analysis report, and a cycle projection report.”

Issue

However, the staff noted that the LRA does not provide the details regarding the software package that will be used. It is not clear to the staff if the “program” being referred to is the Metal Fatigue of Reactor Coolant Pressure Boundary Program or the

software program. It is also not clear to the staff if the software package will be used for cycle counting only or if it will also be used for cycle-based or stress-based fatigue analysis and includes periodic CUF updates.

Request

(a) Clarify, in detail, how the software package selected will be capable of monitoring those transients that are significant to fatigue usage such that the design limit of 1.0 is not exceeded during the period of extended operation, consistent with the recommendations in GALL AMP X.M1.

(b) Clarify how the software package will perform periodic CUF updates, consistent with the recommendations of the "detection of aging effects" program element of GALL AMP X.M1.

(c) Clarify how the software package referenced in LRA Section B.2.3.1 and Commitment No. 42 addresses and resolves the issue associated with NRC RIS 2008-30.

RAI B.2.1.1-1

Background

The applicant's ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program states that the aging management program (AMP) is "an existing program consistent with NUREG-1801, Section XI.M1." GALL AMP XI.M1 recommends the use of American Society of Mechanical Engineers (ASME) Section XI Table IWB-2500-1 to determine the examination of Category B-F and B-J welds. The applicant is currently including applicable portions of the categories B-F and B-J in its Risk Informed Inservice Inspection Program.

Issue

The staff noted that the approval of the risk-informed methodology cannot be assumed for subsequent ten-year intervals.

Request

Clarify how the inspection of Categories B-F and B-J will be implemented as part of the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program during the period of extended operation.