September 11, 2009

Mr. Jon Franke, Vice President
Crystal River Nuclear Plant (NA1B)
ATTN: Supervisor, Licensing & Regulatory
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15760 W. Power Line Street
Crystal River, FL 34428-6708

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT LICENSE RENEWAL APPLICATION (TAC NO. ME0274)

Dear Mr. Franke:

By letter dated December 16, 2008, Florida Power Corporation submitted an application pursuant to Title 10 of the *Code of Federal Regulations* Part 54, to renew the operating license for Crystal River Unit 3 Nuclear Generating Plant (CR-3), for review by the U.S. Nuclear Regulatory Commission (NRC or the staff). 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. Further requests for additional information may be issued in the future.

Items in the enclosure were discussed during the onsite audit conducted the week of July 13, 2009 with Mr. Michael Heath, 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 by telephone at 301-415-3733 or by e-mail at Robert.Kuntz@nrc.gov.

Sincerely,

/RA/

Robert F. Kuntz, Sr. Project Manager
Projects Branch 2
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket No. 50-302

Enclosure:
As stated

cc w/encl: See next page

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DATE	08/26/09	08/26/09	09/09/09	08/11/09

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Letter to Jon Franke from Robert F. Kuntz dated September 11, 2009

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT LICENSE RENEWAL APPLICATION (TAC NO. ME0274)

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Crystal River Unit 3 Nuclear Generating Plant

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REQUEST FOR ADDITIONAL INFORMATION

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

LICENSE RENEWAL APPLICATION

DOCKET NUMBER 50-302

Request for Additional Information (RAI) 3.3.2-38-1

Background:
Accurate identification of material and environment combinations, as described in NUREG 1801, "Generic Aging Lessons Learned (GALL) Report," is necessary to support aging management reviews.

Issue:
License renewal application (LRA) Table 3.3.2-28, "Aging Management Evaluation for Instrument Air Dryers" describes the instrument air dryers as stainless steel with an internal environment of dried air. During the material/environment verification audit walkdown, the U.S. Nuclear Regulatory (NRC) staff noticed that the instrument air dryers appear to be carbon steel rather than stainless steel as described in LRA Table 3.3.2-28. Also, the internal environment of the dryers, in accordance with the vendor manual, contains alumina desiccant. Further review of the applicable drawings and vendor information did not clarify the type of information for this component.

Request:
Provide the documentation to show that the instrument air dryers material is stainless steel and confirm the internal environment or correct the material and environment descriptions in LRA Table 3.3.2-28.

RAI B.2.1-1

Background:
The Crystal River Unit 3 Nuclear Generating Plant (CR-3) LRA Section B.2.1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD" states that the aging management program (AMP) is "an existing program consistent with NUREG-1801, Section XI.M1."

GALL Report, Section XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD" 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. CR-3 is currently using examination Category R-A in accordance with risk-informed methodology approved by the NRC under 10 CFR Part 50, for use during the current ten-year interval for examination of Table IWB-2500-1, Category B-F and B-J welds.

Issue:
The approval of the risk-informed methodology can not be assumed for the subsequent intervals.

ENCLOSURE

Request:
Clarify how the inspection of Categories B-F and B-J will be implemented during the extended period of operations.

RAI B.2.2-1

Background:
The Monitoring and Trending section of the Water Chemistry AMP in the GALL Report, Section XI.M2, includes periodic monitoring and control of known detrimental contaminants in accordance with the Electric Power Research Institute (EPRI) water chemistry guidelines for pressurized water reactors (PWR). EPRI report 1014986, "Pressurized Water Reactor Primary Water Chemistry Guidelines" (2007) provides guidance to monitor silica in the reactor coolant system during startup daily as indicated in Table 3-8.

Issue:
The CR-3 LRA, Appendix B, Section B.2.2 indicates that the applicant's Water Chemistry Program is consistent with the GALL Report Section XI.M2 and does not take any exceptions. It was indicated in the applicant's basis document L08-0601, Water Chemistry AMP, in Table 6.2-1 on Page 12, that periodic monitoring and control of chemistry parameters are delineated in the EPRI Water Chemistry Guidelines. In addition, the applicant's document entitled, "Crystal River Unit 3 Optimized Primary Chemistry Program" indicates that it has adopted daily measurement of silica consistent with the EPRI Water Chemistry Guidelines. The applicant's CH-400 report, "Nuclear Chemistry master Scheduling Program" provides the schedules for the chemistry monitoring. The CH-400 does not provide guidelines for measuring silica in the reactor coolant system during reactor startup, which is included in the EPRI guidelines and the CR-3 Optimized Primary Chemistry Program.

Request:
Provide additional information on the total silica monitoring program during reactor system startup that clarifies the discrepancy between the CR-3 Optimized Primary Chemistry Program and CH-400 monitoring schedule.

RAI B.2.3-1

Background:
In LRA Appendix B, Section B.2.3, the applicant stated that the Reactor Head Closure Studs Program has an enhancement to select an alternate lubricant that is compatible with the fastener material and the contained fluid. The applicant also stated that except for the enhancement the program is consistent with the GALL Report.

NRC Regulatory Guide (RG) 1.65 is one of the technical references for the GALL Report, "Reactor Head Closure Studs Program" (AMP XI.M3) and states the regulatory position that lubricants for the stud bolting are permissible provided they are stable at operating temperatures and are compatible with the bolting and vessel materials and surrounding environment.

Issue:
The staff noted that a molybdenum disulfide based lubricant is used for the reactor head closure studs. It should be clarified whether the lubricant, which has been used, caused detrimental effects on the bolting and vessel materials. In addition, the staff found a need to clarify how the selection of a new lubricant will consider the stability of the lubricant at operating temperatures as recommended in RG 1.65.

Request:
1. Confirm whether operating experience indicates that the currently used lubricant has caused detrimental effects on the bolting materials.
2. Clarify how the selection of a new lubricant will consider the stability of the lubricant at the operating temperatures as recommended in RG 1.65.

RAI B.2.4-1

Define the Section 8.3 “Deviation” process of Procedure EGR-NGGC-0207 for the Boric Acid Corrosion Program including responsibilities, documentation requirements, as well as the review and authorization process.

RAI B.2.4-2

Guidance for the aging management of other nickel-alloy nozzles and penetrations is provided in the aging management review (AMR) line items of Chapter IV, as appropriate.

Commitment 2 of the License Renewal Commitments of Enclosure 2 of the CR-3 application states:

> In accordance with the guidance of NUREG-1801, Rev. 1, regarding aging management of nickel alloy and nickel-clad components susceptible to primary water stress corrosion cracking, CR-3 will comply with applicable NRC Orders and will implement applicable: (1) Bulletins and Generic Letters and (2) staff-accepted industry guidelines.

During the license renewal audit process at CR-3, the applicant’s response to NRC issues related to the aging management of nickel alloy and nickel-clad components susceptible to primary water stress corrosion cracking was that Commitment 2 would answer these issues when a Ni-based Alloy Aging Management Program would be submitted to the NRC, two years prior to the beginning of the license renewal period. However, Commitment 2 does not commit to the submittal of this report within that timeline. Confirm and/or clarify Commitment 2 in the Program Implementation Schedule.

RAI B.2.5-1

Background:
As provided in GALL AMP XI.M11A, the augmented inspection requirements established in First Revised Order EA-03-009 include visual VT-2 and nondestructive examinations. The final rule for Title 10 of the *Code of Federal Regulations* (CFR) Part 50.55a supersedes the Revised Order and requires all licensees to augment their inservice inspection program with ASME Code Case N-729-1. The examination requirements for reactor vessel upper heads are visual

examination, volumetric, and/or surface examination. Personnel performing the visual examination shall be qualified as a VT-2 visual examiner and shall have completed at least four hours of additional training in detection of borated water leakage.

Issue:
The applicant's Inservice Inspection (ISI) Components and Structures Examination Program, Revision 8 states that Code Case N-729-1 will be implemented until Refuel Outage 18 in 2013 in accordance with 10 CFR 50.55a. The applicant also states that the reactor vessel upper head shall receive a visual inspection every other outage starting with the first outage after January 1, 2009, and a volumetric inspection not to exceed every ten calendar years following the initial examination. However, specific schedules for the visual and volumetric inspections and implementation of the additional training for visual examination are not provided.

Request:
Clarify the specific schedules for the visual and volumetric inspections and how the additional training for visual examination will be implemented.

RAI B.2.6-1

Background:
The scope of the Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Aging Management Program in the GALL Report, Section XI.M13 includes consideration of synergistic loss of fracture toughness due to both neutron embrittlement and thermal aging embrittlement. In addition, under the parameters monitored/inspected, the CASS materials identified include those with neutron fluences of greater than 10^{17} n/cm^2 (E>1 MeV) or those that are susceptible to thermal embrittlement.

Issue:
The CR-3 LRA, Appendix B, Section B.2.6 indicates that the applicant's Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel Program is consistent with the GALL Report Section XI.M13 and does not take any exceptions. It was indicated in the applicant's basis document L08-0607, Thermal Aging and Neutron Irradiation Embrittlement of CASS, in Table 6.2.1 on Page 9, that unless the CASS component is subjected to a fluence level of 10^{21} n/cm^2 (E>1 MeV) or higher, the synergistic effect of thermal aging and neutron irradiation embrittlement need not be evaluated. EPRI report 1015395, "Plant Support Engineering: Utility Cast Austenitic Stainless Steel Component Aging Management Inspection Needs" is referenced in regards to this conclusion, but does not provide the basis data that supports this assertion.

Request:
Provide additional information that justifies limiting the synergistic loss of fracture toughness consideration to fluence levels greater than 10^{21} n/cm^2 (E>1 MeV) in lieu of the GALL Report recommended 10^{17} n/cm^2 (E>1 MeV) limit. Describe whether this proposal limit is consistent with other industry work (e.g. EPRI MRP-227, "PWR Reactor Internals Inspection & Evaluation Guidelines").

RAI B.2.6-2

Background:
The detection of aging effects of the Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel Aging Management Program in the GALL Report, Section XI.M13 indicates that the 10-year ISI program include a supplemental inspection for those components that have a fluence of greater than 10^{17} n/cm^2 (E>1 MeV) or are susceptible to thermal embrittlement, unless a component-specific evaluation determines it unnecessary. In addition, the guidance indicates that the inspection technique used should be capable of detecting the critical flaw size with adequate margin, which will be based on service loading conditions and service-degraded material properties.

Issue:
The CR-3 LRA, Appendix B, Section B.2.6 indicates that the applicant's Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel Program is consistent with the GALL Report, Section XI.M13 and does not take any exceptions. It was indicated in the applicant's basis document L08-0607, Thermal Aging and Neutron Irradiation Embrittlement of CASS, in Table 6.2.1 on Page 13, that unless a component specific evaluation is conducted, the susceptible components will be inspected under an augmented inspection program. The inspection technique and acceptance criterion is to be determined by the CR- 3 ISI Program Manager, and that the GALL Report recommendations must be "considered." In the LRA, Section B.2.6, the applicant indicated that the Thermal Aging and Neutron Irradiation Embrittlement of CASS program will be implemented and required inspections completed during the last 10-year ISI interval prior to the period of extended operation. The applicants basis document L08-0607 indicated that the "augmented inspection," which is part of the program will be identified as an augmented inspection in their basis document AI-701, "Administration of the ASME Section XI Inservice Inspection and Inservice Testing Programs." However this document did not provide further basis data on the techniques that will be used during the last 10-year ISI interval and through the extended operation period.

Request:
Provide additional information on the augmented inspection program of the Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel program B.2.6 that indicates how it is consistent with the GALL Report guidance.

RAI B.2.7-1

Background:
In LRA Section B.2.7.1-1, the applicant states that the CR-3 Flow-Accelerated Corrosion (FAC) Program described in EGR-NGGC-0202, "Flow Accelerated Corrosion Monitoring Program," Revision 10, is based on EPRI guidance document NSAC-202L, "Recommendations for an Effective Flow-Accelerated Corrosion Program," Revision 3, dated May 2006.

Issue:
The GALL Report recommends the use of revision 2 of the NSAC-202L.

Request:
Please provide a discussion of the differences between revisions 2 and 3 of the EPRI guidance document NSAC-202L and provide a discussion as to why this is not considered an exception to the GALL Report.

RAI B.2.7-2

Background:
In LRA Section B.2.7, the applicant states that the FAC Program monitors the effect of FAC on the intended function of piping and components by measuring wall thickness. It was further stated that selection and prioritization of components to be inspected consider NSAC-202L, using the following criteria: CHECWORKS model predictions, trending, consequences of failure, engineering judgment, and plant and industry operating experience events.

Issue:
The LRA did not contain information regarding the accuracy of the FAC Program in predicting FAC degradation in components.

Request:
Please provide a sample list of components for which wall thinning is predicted and measured by ultrasonic testing or other methods in order to assess the accuracy of the FAC predictions from CHECWORKS. This list should also include the initial wall thickness (nominal), current (measured) wall thickness, and a comparison of the measured wall thickness to the thickness predicted by the CHECWORKS FAC model.

RAI B.2.8-1

Background:
GALL AMP XI.M18, “Bolting Integrity,” states that the program relies on the recommendations for a comprehensive bolting integrity program as delineated in NUREG-1339, which is the resolution of Generic Safety Issue 29 on bolting degradation or failure, and relevant industry recommendations such as EPRI TR-104213. NUREG-1339 states that facts from some service failures and from laboratory examinations clearly show that molybdenum disulfide is a potential contributor to stress corrosion cracking (SCC).

In LRA Appendix B, Section B.2.8, the applicant stated that the Bolting Integrity Program has an enhancement to identify and remove instances where molybdenum disulfide lubricant is allowed for use in bolting applications in specific procedures and to add a specific prohibition against use of molybdenum disulfide lubricants in the CR-3 procedure for bolted connections. The staff noted that this enhancement is in agreement with the NRC staff findings summarized in the “Conclusions” section of NUREG-1339 and Generic Letter 91-17.

Issue:
It should be clarified whether the molybdenum disulfide lubricant, which was used previously or is used currently, has caused detrimental effects on the bolting materials or not.

Request:
Describe what lubricants, which have been used in the applicant's bolting integrity program, are based on molybdenum disulfide. Clarify whether the lubricants have caused detrimental effects on the bolting materials or not. Provide justification for the applicant's evaluation of the detrimental effects including relevant operating experience.

RAI B.2.8-2

Background:
In LRA Appendix B, Section B.2.8, the applicant stated that the Bolting Integrity Program has enhancements for the program elements, Parameters Monitored/Inspected and Detection of Aging Effects. The applicant stated that the enhancements are to perform periodic ultrasonic examination of a representative sample of bolting identified as potentially having actual yield strength > 150 ksi. During the AMP audit, the applicant agreed that the yield strength criterion should be corrected from "> 150 ksi" to "≥ 150 ksi" in a manner consistent with the recommendation of NUREG-1339.

Issue:
GALL AMP XI.M18 indicates that high strength low alloy steel with the actual yield strength ≥ 150 ksi may be subject to SCC. The "Conclusions" section of NUREG-1339 recommends that the yield strength criteria for categorization of material's SCC susceptibility should be based on actual measured yield strength (or yield strength determined by conversion of measured hardness values), but not on the specified minimum yield strength in material specifications. One method to verify actual yield strength is to refer to the test data of Certified Material Test Reports. Based on foregoing information, the staff found a need to clarify the applicant's approach related to this topic.

The staff noted that on-site program document, L07-0372, "License Renewal Aging Management Review for the Reactor Building," Revision 2, Section 6.3.14.14, "Threaded Fasteners" addressed a calculation of the maximum yield strength using the specified minimum yield strength and specified maximum and minimum tensile strength values as described on Page 67 in relation with categorization of materials in terms of SCC susceptibility. This calculation suggests that the applicant might use specified yield and/or tensile strength values to calculate yield strength and the calculated yield strength might be used as input for SCC susceptibility categorization of materials.

Request:
Describe how the yield strength of the bolting materials will be determined as input for the yield strength criterion to categorize the materials in terms of SCC susceptibility (For example, based on actual measured yield strength, conversion of actual measured hardness, Certified Material Test Reports or specified values of yield strength, tensile strength and/or hardness in material specifications).

RAI B.2.8-3

Background:
LRA Appendix B, Section B.2.8 and onsite program documentation indicated that the program enhancements will use "EPRI-5067" (EPRI NP-5067, Volumes 1 and 2) as guidance for torquing and closure requirements and as a basis to develop a centralized procedure for joint leak tightness and pre-installation inspections.

Issue:
GALL AMP XI.M18, "Bolting Integrity," refers to EPRI NP-5769 and EPRI TR-104213 as guidance for industry recommendations. The staff also noted that EPRI NP-5769 Volume 1, Section 2 (Page 2-8) states that EPRI NP-5067 "will serve as a repository of useful information learned from EPRI experimental and analysis programs and will give the utility industry guideline." EPRI NP-5769 also states that "It is believed that the bolting reference manuals [NP-5067, Volumes 1 and 2] will satisfy the industry's need for guidance in this area."

However, EPRI TR-104213, Section 1.1 indicates that the development objectives for TR-104213 were to update and consolidate the existing information (including NP-5067 and NP-6316) into a single document and to provide additional information necessary to allow a seamless integration of the material. The staff also noted that NUREG-1339 takes some exceptions for safety related bolting to EPRI NP-5769 such as yield strength criteria for categorization of materials in terms of SCC susceptibility.

Request:
Clarify whether EPRI NP-5769 with the exceptions noted in NUREG-1339 and EPRI TR-104213 will be considered and used as industry recommendations for the applicant's enhancements as well as EPRI NP-5067 that the applicant is currently planning to use. If necessary, describe how the final safety analysis report (FSAR) supplement in LRA Appendix A, Section A.1.1.8 will be revised.

RAI B.2.9-1

Background:
The applicant stated in Section B.2.9 of the LRA that its Steam Generator Tube Integrity program is an existing program that is consistent with the recommendations of the GALL Report, Section XI.M19, "Steam Generator Tube Integrity." It further stated that its program meets the intent of NEI 97-06 as recommended by the GALL Report.

Issue:
CR-3 program document, Nuclear Generation Group, L08-0634, describes the steam generator tube integrity aging management program. The aging management program indicates the program is consistent with the GALL Report with no exceptions. The GALL Report indicates to effectively manage the effects of aging that a licensee must, in part, implement the steam generator degradation management program described in Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines." Revision 1 of these guidelines is referenced in the GALL Report, whereas the application references revision 2 of NEI 97-06.

Request:
Please justify the use of Revision 2 of NEI 97-06.

RAI B.2.9-2

Background:
The applicant indicated that the steam generator tube integrity program at CR-3 meets the intent of NEI 97-06 as recommended by the GALL Report.

Issue:
During the review of the CR-3 steam generator tube integrity program, numerous potential discrepancies within the same procedure, between different procedures, between the CR-3 procedures and various industry guidelines (referenced in NEI 97-06), and between the procedures and the technical specifications were identified. These issues are discussed below:

- In Section 3.1.1 of EGR-NGGC-0208, accident-induced leakage is defined, in part, as the primary-to-secondary leakage that occurs during postulated accidents when tube structural integrity is assumed. Although this definition is consistent with NEI 97-06, accident induced leakage should simply include all primary-to-secondary leakage prior to the accident and any additional leakage induced as a result of the accident (regardless of whether structural integrity is maintained). The technical specifications and bases (that were provided in the license amendment that adopted TSTF-449) do not include the "exception" that accident induced leakage does not include leakage as a result of loss of structural integrity.

- In Section 3.1.4 of EGR-NGGC-0208, it is indicated that condition monitoring is performed at the conclusion of each operating cycle. Although this frequency is consistent with NEI 97-06, it is not consistent with your technical specifications (5.6.2.10.a). The staff notes that this definition is also inconsistent with a similar definition in Section 3.3.3 of SP-305. The staff further notes that the technical specifications require condition monitoring following an outage in which inspections are performed or tubes are plugged or repaired.

- In Section 8.2.3 of EGR-NGGC-0208, it was indicated that changes in design parameters shall be assessed and included in the assessment of structural limits if they result in a primary-to-secondary pressure difference that is greater than the established value for normal operation by more than 50 pounds per square inch. Since the technical specifications require maintaining a margin of 3 against the normal operating differential pressure, it is not clear that the technical specifications would be satisfied if this procedural step were followed.

- In Section 8.4.3 of EGR-NGGC-0208, it is indicated that the pressure and temperature conditions used in the determination of the primary-to-secondary leakage rate shall be consistent with the guidance in the industry's primary to secondary leak guidelines. The acceptability of the leakage rate limits in your technical specifications and licensing basis were evaluated for a specific set of conditions. It is not clear that the industry's primary-to-secondary leakage guidelines require using the pressure and temperature conditions

assumed in the design and licensing basis. This could result in calculating a leakage rate under one set of conditions and comparing it to limits that assumed a different set of conditions, which would not be appropriate.

- Section 9.5.6 of EGR-NGGC-0208 indicates that if the condition monitoring results are not as expected and/or growth rates are larger than expected, one or more tubes may fail to satisfy the performance criteria may fail to meet the performance criteria prior to the next inspection. The procedure then indicates remedial actions (e.g., limiting the length of inspection interval) can be taken. To ensure compliance with the technical specifications, if one or more tubes may fail to meet the performance criteria, remedial actions must be taken.

- Section 9.6.4.2 of EGR-NGGC-0208 indicates (appropriately) that leakage from all sources (e.g., plugs) must be assessed. Section 9.6.4.4 of EGR-NGGC-0208 indicated that if projected degradation is not calculated to penetrate the tube wall and projected worst case end-of-cycle degradation is not calculated to tear through the wall at accident pressure differentials there is no projected operational assessment leakage. Since primary to secondary leakage may occur as a result of leakage past plugs without penetrating the wall or tearing through the wall, it appears that Section 9.6.4.4 of EGR-NGGC-0208 could lead to a wrong conclusion regarding compliance with the technical specifications.

- Section 9.6.8 of EGR-NGGC-0208 indicates that if one or more tubes fail to meet the structural or accident induced leakage performance criteria prior to the next outage a condition report should be generated. It is not clear from this section (unlike other sections) that leakage from all sources would be summed to ensure that the accident induced leakage performance criteria is satisfied.

- Section 9.6.9 of EGR-NGGC-0208 indicates that if leakage causes a forced outage, then a condition report would be generated to ensure a root cause analysis for failing to meet the criteria is performed. It is not clear why a root cause evaluation would only be performed when the “criteria” (presumably the operating leakage performance criteria) is not satisfied. The industry’s integrity assessment guidelines would normally require an evaluation when the leakage exceeds what is predicted (i.e., the previous operational assessment did not bound what was observed).

- Section 9.9.10 of EGR-NGGC-0208 indicates, in part, that if visual damage of the tubes is detected or considered likely that you should “determine and document the need for tube integrity assessment.” This appears contradictory to the requirements of the Integrity Assessment guidelines (referenced by NEI 97-06) which indicate tube integrity shall be evaluated when tube damage is detected or considered likely.

- Section 3.3.1 of SP-305 defines alternate repair criteria different than Section 3.1.2 of EGR-NGGC-0208. The latter indicates that alternate repair criteria can be implemented if approved by the NRC (which is correct) while the former does not have this stipulation.

- Section 3.6.5 of SP-305 indicates that welded plugs showing signs of leakage are unacceptable and shall be evaluated for repair, replacement, or acceptable for continued service. It is not clear how an unacceptable plug could be considered “acceptable” for continued service.

- Section 8.4.4 indicates that the measurement and detection methods and associated actions shall adhere to the intent of the industry’s primary-to-secondary leak guidelines. The reason for not fully implementing the primary-to-secondary leak guidelines (e.g., the sampling frequency requirements) is not clear.

- Section 3.3.5 through 3.3.9 and Section 3.3.21 of SP-305, terms are defined that were previously defined in technical specifications. Although some of these are consistent with definitions in the EPRI guidelines, they are no longer used in the technical specifications. In addition, not all of these terms appear to be used in the guidelines or in your procedures.

- Section 3.6.3 of SP-305 indicates that the steam generators shall be determined operable after plugging or repairing all tubes exceeding the plugging/repair limit. This was appropriate under the previous version of your technical specifications; however, the current version of your technical specifications also requires tube integrity to be maintained. As a result, the condition cited in your procedures is not sufficient for ensuring operability.

- Section 4.2.3 of SP-305 indicates that the inspection scope should be expanded, in part, per the technical specifications. The technical specifications no longer have specific guidance on expanding the number of tubes inspected.

- Section 3.6.1 of SP-305 lists five criteria for when tubes can be returned to service. At least one of the criteria must be satisfied for the tubes to be returned to service. One of these criteria is that the tubes have adequate margin against burst under normal full power and postulated accident conditions. This criterion must be satisfied for a tube to be returned to service (regardless of whether the tube meets any one of the other four criteria). Since the technical specifications require all tubes have adequate structural integrity, which includes burst and collapse, under a variety of operating conditions and that the steam generator has adequate leakage integrity, it is not clear whether these conditions are fully appropriate.

- Section 3.6.2 of SP-305 indicates that the tubes not inspected during the periodic inspection shall be considered acceptable to return to service provided the minimum inspection percentage including any required sample expansion has been met. Since the technical specifications require tube integrity to be maintained, it does not appear appropriate to just assume that because the minimum sample size was met that tube integrity would be maintained (e.g., growth rate larger than anticipated in operational assessment).

- Sections 3.6.6 and 3.6.7 of SP-305 addresses, in part, implementation of the accident induced leakage performance criteria. In Section 3.6.6 it indicates, in part, that the limit is one gallon per minute minus 150 gallons per day per steam generator. In Section 3.6.7, it indicates that the criterion is 0.699 gallons per minute. These numbers do not appear consistent. The staff also notes that leakage during normal operation may increase during design basis accidents and, therefore, simply subtracting the normal operating leakage limit of 150 gallons per day from the one gallon per minute steam line break leakage limit may not be appropriate.

- Section 4.2.3 of SP-305 indicates that eddy current data should be analyzed in accordance with the EPRI Pressurized Water Reactor Steam Generator Examination Guidelines. These guidelines do not really have sufficient information to analyze eddy current data. There are eddy current data analysis guidelines that appear to be used at CR-3, but these are not referenced in this procedure. In addition, these guidelines do not appear to be current since they reference outdated versions of the Pressurized Water Reactor Steam Generator Examination Guidelines. NEI 97-06 requires licensees to modify their steam generator programs when new guidance is issued (within the time limits specified in the transmittal letter for the new guidelines).

- Section 4.3.1 of SP-305 implies that the only tubes that require plugging are defective tubes, tubes identified during a bubble test, or tubes that exceed the repair criteria. Since there may be instances where these conditions are not satisfied but yet the tubes are required to be plugged to ensure tube integrity is maintained, it is not clear that your procedures will ensure compliance with your technical specifications.

- Section 4.3.1 of CP-152 directs the operator to go to Section 4.8 if RM-A12 is unavailable to quantify primary-to-secondary leakage and the leak rate is greater than or equal to 75 gallons per day. Section 4.8 does not appear to address the condition when RM-A12 is unavailable and therefore the plant would not be required to be in Mode 3 within 6 hours. This is a mandatory item of the PWR Primary-to-Secondary Leak Guidelines (referenced in NEI 97-06). Similarly, Section 4.3.3 directs the operator to go to Section 4.7 under specific conditions; however, Section 4.7 does not appear to address the condition when RM-A12 is unavailable.

- The PWR Primary-to-Secondary Leak Guidelines (referenced in NEI 97-06) requires increased grab sampling frequency when Action Level 1 is reached. CP-152 does not appear to require increased grab sampling frequency when this Action Level is reached rather it appears to require the same sampling as was required when the leakage was slightly less than Action Level 1 levels.

- The PWR Primary-to-Secondary Leak Guidelines requires that the method used to monitor the rate of increase in leakage for Action Level 2 shall be specified. In Section 4.7.1 of CP-152, it appears that a specific method is chosen and it is tied to a specific monitor (RM-A12). This section of CP-152 is also referenced when RM-A12 is unavailable. As a result, the method for monitoring the rate of increase does not appear to be specified in cases where RM-A12 is unavailable.

- The Steam Generator Integrity Assessment Guidelines (referenced in NEI 97-06) indicate that the degradation assessment shall include secondary side considerations such as foreign object search and retrieval. The degradation assessment for the replacement steam generators (June 2009) does not appear to address secondary side activities. Loose parts (foreign objects) have been found on the secondary side of replacement steam generators.

- The Steam Generator Integrity Assessment Guidelines indicate, in part, that the limiting structural integrity performance criteria and the appropriate loading conditions for the degradation of interest shall be identified in the degradation assessment. In addition, these guidelines indicate that the condition monitoring and operational assessment limits for all existing and potential degradation mechanisms and the appropriate measurement parameter for each degradation mechanism be identified in the degradation assessment. This information does not appear to be in the most recent degradation assessment for your current steam generator.

- The Steam Generator Integrity Assessment Guidelines indicate, in part, that projections of leakage at normal operating conditions shall be performed in the operational assessment. This information does not appear to be in the operational assessment.

- In Section 3.1 titled, “From NEI-06” of the applicant’s document EGR-NGCC-0208, some definitions in the NEI 97-6 Revision 2 document “Steam generator program guidelines” do not appear, such as “collapse” or “repair methods”. Furthermore, the applicant gave definitions that appear to have come from the NEI 97-06 guidelines but are not defined in this document, such as “Degradation-specific repair criteria”, “Faulted”, etc…

- NEI 97-06 is referenced in the GALL Report for the steam generator tube integrity program and constitutes the basis of the applicant’s procedures to manage the steam generator tube integrity program. These gaps could induce some confusion in the procedures used by the applicant in order to follow the NEI guidelines, particularly for inspections, and thus manage the aging effects of steam generator tubes.

- In Section 5.3.6 of the document “Eddy current data analysis guidelines for the once-through steam generator inservice inspection”, the applicant stated that “a recent site-specific data analysis performance demonstration (SSPD) performed at a similar design plant may be accepted in lieu of all or portion of the CR-3 SSPD”. The EPRI document “Examination guidelines” specifies in Section G.4.3.1.1 relative to SSPD examinations that “a written test evaluates knowledge of the specific plant SG history and conditions, examinations techniques, expected damage mechanisms, and unique challenges to examination process”.

- It is not obvious that another plant would have identical history, conditions, and challenges to the examination process to CR-3 to permit use of the SSPD from the other plant.

Request:
Given the number of potential discrepancies, discuss your plans to perform a comprehensive review of your steam generator program to ensure the procedures are internally consistent, will ensure compliance with the technical specifications, and are consistent with NEI 97-06.

RAI B.2.9-3

Background:
The applicant stated in Section B.2.9 of the LRA that its Steam Generator Tube Integrity program is an existing program that is consistent with the recommendations of the GALL Report, Section XI.M19, "Steam Generator Tube Integrity." It further stated that its program meets the intent of NEI 97-06 as recommended by the GALL Report.

Issue:
During the review of CR-3 program document, EGR-NGGC-0208, a definition of faulted was provided which indicates it only includes secondary side depressurizations. The reason for restricting this definition to only secondary side depressurizations is not clear. Since it is not evident how this definition is used in the steam generator program, unnecessarily restricting this definition to one class of design basis accidents may be inappropriate (and result in not meeting regulatory requirements).

Request:
Please discuss how the term "faulted" is used in the program and, if it is used, discuss the reason for only limiting the definition to secondary side depressurizations.

RAI B.2.9-4

Background:
The applicant stated in Section B.2.9 of the LRA that its Steam Generator Tube Integrity program is an existing program that is consistent with the recommendations of the GALL Report, Section XI.M19, "Steam Generator Tube Integrity." It further stated that its program meets the intent of NEI 97-06 as recommended by the GALL Report.

Issue:
Section 9.7 of EGR-NGGC-0208 indicates tube plugging will be performed to industry standards. This reference is not clear. In addition, the Inservice Inspection Program Plan (dated December 21, 2007, 3F11207-04) does not appear to address steam generators. 10 CFR 50.55a requires, in part, that Section XI of the ASME Code be followed with certain exceptions. Section XI of the ASME Code addresses, in part, steam generator tube plugging and inspections and performance of non-destructive examinations. For steam generators, there is an exception in 10 CFR 50.55a that if the technical specification surveillance requirements are different than those in the ASME Code then the technical specifications govern. Since the technical specifications do not address all aspects of steam generator inspection and maintenance (e.g., plugging), the lack of reference to the ASME Code in the steam generator program raises questions on whether the requirements of the ASME Code are being met.

Request:
Please confirm that the ASME Code requirements pertaining to steam generator tube plugging, sleeving, and non-destructive examination are being followed (for those instances where there is no conflict with the specific requirements in the technical specifications).

RAI B.2.9-5

Background:
In Sections 4.1.1 and 4.1.2 of the document SP-305 “Once-Through Steam Generator Inservice inspection” relative to visual inspection of tube plugs, the applicant used the expression “PGN approved procedures” without referencing these procedures in this document.

Issue:
It isn’t clear which procedures should be applied according to the document SP-305.

Request:
Please identify the specific PGN procedures discussed in SP-305.

RAI B.2.10-1

Background:
The applicant states that its LRA AMP Open Cycle Cooling Water System (B.2.10) is consistent with the GALL AMP, Open Cycle Cooling Water System (XI.M20). In its audit of program Element 4 (acceptance criteria), the staff identified a potential inconsistency between the LRA AMP and the GALL AMP.

Issue:
LRA AMP program Element 4, detection of aging effects, contains inspections techniques typically employed by Generic Letter 89-13 and which are consistent with the corresponding element in the GALL AMP. Section 7.5 of the applicant’s document L08-0613, “License Renewal Aging Management Program Description of the Open Cycle Cooling Water System Program” indicates that, in addition to its normal uses, the LRA AMP will be used to detect selective leaching. The selective leaching AMP utilizes both visual inspection techniques and hardness/scratch tests to identify selective leaching. It is not clear to the staff that the LRA AMP will adequately detect selective leaching unless it is enhanced to include some form of hardness testing.

Request:
Justify the exclusion of hardness testing for the identification of selective leaching or justify how this aging effect may be identified using the inspection techniques already specified.

RAI B.2.10-2

Background:
The applicant states that its LRA AMP, Open Cycle Cooling Water System (B.2.10), is consistent with the GALL AMP, Open Cycle Cooling Water System (XI.M20). In its audit of program Element 6 (acceptance criteria), the staff identified a potential inconsistency between the LRA AMP and the GALL AMP.

Issue:
Section A.1.2.3.6 of the Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants (SRP-LR) states that the program element "acceptance criteria" should contain information concerning the acceptance criteria against which the need for corrective action will be measured. This section of the SRP-LR also states that the acceptance criteria should consist of numerical values or methods by which they are determined. The staff notes that this information is absent from this section of the LRA AMP. In order for the staff to evaluate the consistency of this LRA program element with the corresponding GALL Report program element, it is necessary that the applicant provide this information in the LRA AMP.

Request:
Please provide acceptance criteria as indicated in the SRP-LR.

RAI B.2.10-3

Background:
The applicant states that its LRA AMP, Open Cycle Cooling Water System (B.2.10), is consistent with the GALL AMP, Open Cycle Cooling Water System (XI.M20). In its audit of program Element 10 (operating experience), the staff identified a potential deficiency in the LRA AMP.

Issue:
Assessment number 149036, Observation number JL06, Service Water Reliability, GL 89-13 Program Assessment indicates that the concrete lining is missing from some sections of the concrete lined, cast iron, 48 inch piping. This assessment also indicates that the sections of the piping from which the concrete is missing are not being monitored for loss of material. Since the GALL AMP is designed to manage aging for lined piping and since corrosion of unlined piping is expected to be greater than for lined piping, it is not clear that the LRA AMP will adequately manage aging for the areas of pipe without concrete lining without additional inspection or control measures.

Request:
Please justify how the LRA program will adequately manage aging in the unlined sections of piping or propose enhancements to the program which consider these piping sections.

RAI B.2.11-1

Background:
The CR-3 LRA proposes an exception to the GALL Report recommendation by not subjecting the closed-cycle cooling water pumps to a formal testing program. The LRA states that the ability of the systems to maintain adequate flow rates and heat transfer is verified on an ongoing basis by routine operation of the systems.

Issue:
The GALL Report recommends monitoring the following pump parameters: flow, discharge, and suction pressures as a part of system and component evaluation.

Request:
Please provide details on how the system's ability to maintain flow rates and heat transfer is to ensure without subjecting the cooling water pumps to a formal testing program. Particularly, (1) state the surrogate parameters, if any, that are being monitored as a part of system health evaluation if the GALL Report recommended parameters are not monitored, and (2) provide justification as to how the surrogated parameters could provide equivalent protection to the cooling water pumps.

RAI B.2.11-2

Background:
The CR-3 LRA proposes an exception to the GALL Report recommendation by not subjecting the closed-cycle cooling heat exchangers to a formal performance monitoring program. The LRA states that the thermal and hydraulic performance of the systems is verified on an ongoing basis by routine operation of the systems.

Issues:
The GALL Report recommends monitoring the following heat exchanger parameters: flow, inlet and outlet temperatures, and differential pressure as a part of system and component evaluation.

Request:
Provide details on how the systems' ability to maintain flow rates and heat transfer is to ensure without subjecting the cooling water heat exchangers to a formal testing program. Particularly, please state the surrogate parameters that are being monitored as a part of system health evaluation if the GALL Report recommended parameters are not monitored and provide justification as to how the surrogated parameters could provide equivalent protection to the cooling water heat exchangers.

RAI B.2.11-3

Background:
The CR-3 LRA states that the corrosion inhibitor concentrations stayed with the EPRI Closed Cooling Water Chemistry Guidelines with no reference to a specific version of the report. The GALL Report recommends following the guidance in EPRI TR-109376, "Closed Cooling Water Chemistry Guideline" dated 1997.

Issue:
The applicant references EPRI TR 1007820 "Closed Cooling Water Chemistry Guideline," Revision 1 date 2004, in its plant procedures CP-160 and CH-400 as the basis for the action level limits, action responses, and sampling frequency for control and diagnostic parameters.

Request:
Please provide (1) a comparison between the CR-3 Closed Cycle Cooling Water System Program control/diagnostic parameters, associated limits, and sampling frequency to those found in the EPRI 1997 and 2004 guidelines, and (2) a justification as to why an exception to the GALL Report was not taken in the LRA if CR-3 opts to follow the 2004 version of the EPRI guideline.

RAI B.2.13-1

Background:
The CR-3 LRA proposes an exception to GALL AMP XI.M26 on the Halon fire suppression system testing frequency. The GALL Report recommends a 6-month inspection and testing frequency of the Halon system. The applicant proposes to test the Halon system once every 18 months during the refueling outage. The applicant states that the exception is acceptable because the Halon system is located in a conditioned air environment.

Issue:
The functional testing frequency of the Halon system does not meet the GALL recommended frequency.

Request:
Please provide:

1. reason(s) based on the plant operating experience and other relevant details (e.g., the manner in which the air is conditioned and/or filtered to remove corrosive substances, etc.) to justify the extended functional testing frequency for the Halon system.
2. details on any corrective/compensatory action(s) taken in the event when the air condition is out of service.
3. the Code of Record year for the NFPA 12A "Standards on Halon 1301 Fire Extinguishing Systems" for CR-3.

RAI B.2.13-2

Background:
The CR-3 LRA proposes an exception to GALL AMP XI.M26 on the frequency of visual inspection of walls, ceilings and floors commensurate with the safety significance of the structure and its condition but not to exceed 10 years. The GALL Report recommends that visual inspections of fire barrier walls, ceilings, and floors be performed at least once every refueling outage.

Issues:
The applicant's proposed fire barrier visual inspection frequency does not meet the recommended frequency given in the GALL Report.

Request:
Please provide the basis for exceeding the GALL Report recommended frequency of inspecting the walls, ceilings, and floors.

RAI B.2.14-1

Background:
The CR-3 LRA proposes an enhancement to "either replace the sprinkler heads prior to their 50-year service or perform field service testing of representative samples from one or more sample areas by a recognized testing laboratory. Subsequent test intervals will be based on test results." The GALL Report recommends following NFPA 25 "Inspection, Testing, and Maintenance of Water-Based Fire Protection Systems" guidelines when replacing and or testing the sprinkler heads.

Issues:
The GALL Report recommends the subsequent testing be performed every 10 years following the initial field service testing. The 10-year testing interval is also referenced in NFPA 25 date 2002, Section 5.3.1.1.1. The applicant states the "subsequent test intervals will be based on test results."

Request:
Please provide reason(s) why it is appropriate to rely on the initial test results to schedule the subsequent sprinkler head testing instead of following the GALL Report recommendation of conducting the subsequent sprinkler head testing at a 10-year interval following the initial testing.

RAI B.2.15-1

Background:
As described in GALL AMP XI.M29, the effects of corrosion of the aboveground steel tanks are detectable by (1) periodic system walkdowns during each outage to monitor degradation of the external coating on the tank surface and the sealant at the metal-concrete interface, and (2) thickness measurement of the tank bottom to assess the underground surface condition.

Issue:
By letter dated December 15, 2008, the licensee, in commitment number 10, commits to implementing this program prior to the period of extended operation. The applicant states that tank bottom surface thickness measurement and internal visual inspections will be performed using the planned preventive maintenance activities. Although no additional thickness measurements are identified, frequency of the thickness measurements will be based on the findings of visual inspections performed. It is not clear how, and the frequency of, tank bottom thickness measurement and internal visual inspection will adequately manage the aging effects of the tanks to ensure their intended function will be maintained during the extended period of operation.

Request:
1. Clarify how internal visual inspections are adequate to exclude corrosion of underground external surface if no additional thickness measurements are performed.
2. Clarify and justify the frequency of tank bottom thickness measurement and internal visual inspection to be performed under the preventive maintenance program will adequately detect and monitor the effects of corrosion of the tank bottom surface.

RAI B.2.16.-1

Background:
In LRA Section B.2.16.2-2, the applicant takes exception to the GALL Report in that, the diesel fire pump fuel storage tank FST-2A and FST-2B are not periodically drained of water, but rather, bottom sampling of the tanks is performed quarterly to determine water buildup in the tank bottom.

Corrective actions were taken in 2009 to address an increasing trend of particulates in tanks FST-2A and FST-2B. The actions taken to address the increase in particulates included flushing, cleaning and refilling of the tanks. It was further stated in nonconformance reports (NCR) 313507 and 309256 that an ultrasonic test (UT) inspection will be performed to evaluate tank conditions and a work activity will be developed to periodically inspect the internal surfaces of the tanks.

Issue:
The LRA does not provide a clear justification for how periodic sampling for water is equivalent to the aging management program described in the GALL Report. Additionally, based on records reviewed, it is not clear whether the cause(s) of the increased particulates is or is not related to the current method of sampling.

Request:
Justify how periodic sampling for water is equivalent to the AMP described in the GALL Report. Additionally, discuss the cause(s) of the 2009 tank particulate issue and whether these are related to the tank sampling process currently employed for FST-2A and FST-2B.

RAI B.2.16 -2

Background:
After the issuance of Revision 1 of the GALL Report, the NRC has issued Information Notice (IN) 2009-02, "Biodiesel in Fuel Oil Could Adversely Impact Diesel Engine Performance." This IN discusses potential issues that may occur with the use of B5 blend fuel oil, such as: suspended water particles, biodegradation of B5, material incompatibility, etc.

Issue:
The LRA did not provide information discussing the concerns of IN 2009-02 and the acceptable or unacceptable use of biodiesel at CR-3.

Request:
Provide a summary of the actions that were taken to determine the impact of IN 2009-02 and the use of biodiesel fuel oil at CR-3. If actions have not been taken yet, describe the actions that CR-3 will take to determine the impact of IN 2009-02 and the acceptable or unacceptable use of biodiesel.

If biodiesel is currently being used at CR-3, describe any problems that CR-3 encountered with the use of biodiesel and the associated corrective actions to prevent reoccurrence in the future.

If biodiesel has been determined to not be acceptable for use at CR-3, describe the actions taken and/or will be taken to prevent its addition into fuel oil supply. Also describe actions that will be taken if it is determined that biodiesel has been added into the fuel oil supply.

RAI B.2.19-1

Background:
The CR-3 LRA Section B2.19, Selective Leaching of Materials Program, states that it is consistent with GALL AMP XI.M33, which includes the AMP's 10 elements (with one exception).

Issue:
GALL AMP XI.M33, Operating Experience element, states that the elements that comprise these one-time inspections (e.g., the scope of the inspections and inspection techniques) are consistent with industry practice and staff expectations. Industry has identified a number of instances attributed to selective leaching that may be applicable to the CR-3 AMP. LRA Section B.2.19 addresses CR-3 plant specific operating experience and Calculation No. L08-0625, "License Renewal Aging Management Program Description for Selective Leaching of Materials Program," Revision 1, identifies experience at other Progress Energy plants, but does not address other industry experience and practices beyond Progress Energy-specific examples for the staff to evaluate the acceptability of the AMP.

Request:
For AMP B.2.19 provide additional description of the industry operating experience searched and reviewed and how it will be implemented or utilized for the basis and actions of the CR-3 Selective Leaching AMP. Also provide specifics as to data bases, sources and documents searched, key search terms, and time periods.

RAI B.2.19-2

Background:
The CR- 3 LRA Section B.2.19, Selective Leaching of Materials Program, states that the program is consistent with GALL AMP XI.M33, which includes the AMP 10 elements (with one exception).

Issue:
GALL AMP XI.M33, Program Description, identifies brackish water as an environment that the AMP is intended to address. The applicant's LRA Section B.2.19 and the supporting calculation number L08-0625, "License Renewal Aging Management Program Description for Selective Leaching of Materials Program," Revision 1, identify a number of water environments as being applicable to this AMP, but does not specifically identify brackish water as one of the environments.

Request:
For AMP B.2.19, verify that the identified environments do or do not envelope a brackish water environment; and, if not, provide the technical basis for not considering brackish water.

RAI B.2.20-1

Background:
The LRA states that the Buried Piping and Tanks Inspection Program (B.2.20) is consistent with the GALL AMP, Buried Piping and Tanks Inspection (XI.M34). In its audit of program Element 4 (detection of aging effects), the staff identified a potential inconsistency between the LRA AMP and the GALL AMP.

Issue:
Section A.1.2.3.4 of the SRP-LR states that the program element "detection of aging effects" should contain information concerning the frequency, extent, sample size and methods used to detect aging. The staff notes that much of this information is absent from this section of the LRA AMP. In order for the staff to evaluate the consistency of this LRA program element with the corresponding GALL Report program element, it is necessary that the applicant provide additional information concerning the program for detection of aging effects.

Request:
Please provide additional details of the proposed inspection program.

RAI B.2.21-1

Background:
In the CR3 LRA Section B.2.21, "One-Time Inspection of ASME Code Class 1 Small-Bore Piping," the applicant states that the program is consistent with the program elements in GALL AMP XI.M35, "One-Time Inspection of ASME Code Class 1 Small Bore Piping," including program Element 5, "Monitoring and Trending."

Issue:
During an onsite audit review, the staff noted that the applicant does not have specific information regarding the subject small bore piping weld population, and inspection sampling location and size. This information is needed to evaluate consistency with the GALL Report program element.

Request:
Provide quantitative description of the weld population, including total number of Class 1 small bore welds less than 4 inch NPS, the number of socket welds, and the number of welds two inch or less. Describe your methodology considered in determining the one-time inspection sampling size and locations, and how the program addresses factors such as susceptibility, risk, and operating experience.

RAI B.2.21-2

Background:
In the CR-3 LRA Section B.2.21, "One-Time Inspection of ASME Code Class 1 Small-Bore Piping," the applicant stated that the program is consistent with the program elements in GALL AMP XI.M35, "One-Time Inspection of ASME Code Class 1 Small Bore Piping," which recommends one-time volumetric inspection of small bore piping.

Issue:
The CR-3 application stated that its small bore socket welds will be excluded from the volumetric examination. No additional information was provided regarding its technical basis.

Request:
Please explain how the aging effects of SCC and fatigue in socket welds are addressed. Provide your supporting program information and technical basis.

RAI B.2.22-1

Background:
The GALL Report states that AMP XI.M36, "External Surfaces Monitoring," manages aging of "steel" components (defined in the GALL Report as carbon steel, alloy steel, etc., and "does not include stainless steel") for loss of material, through visual examinations to identify indications of:

- corrosion and material wastage (loss of material)
- leakage from or onto external surfaces
- worn, flaking, or oxide-coated surfaces
- corrosion stains on thermal insulation
- protective coating degradation (cracking and flaking)

The scope of this AMP indicates that it is intended to identify visible rust or rust byproducts (e.g., discoloration or coating degradation) such that loss of material caused by corrosion can be detectable prior to any loss of intended function. This AMP does not identify other aging effects beyond loss of material.

CR-3 LRA Section B.2.22 identifies the plant's External Surfaces Monitoring Program as an existing program that, following an enhancement, will be consistent with the GALL Report. The supporting calculation, L08-0635, states that procedures will be enhanced to assure proper monitoring. L08-0635 states that the program will monitor "loss of material, hardening and loss of strength of elastomers, and reduction of heat transfer caused by fouling." The LRA states that for elastomers, "…the program will include inspection attributes for aging effects that could reasonably be detected through visual inspection. Attributes that require physical manipulation are not included in the program."

Issue:
The intent of GALL AMP XI.M36 is to monitor the aging effects of carbon steel and ferrous materials through visual inspections to identify corrosion, corrosion products, and coating degradation. The program is not intended to monitor reduction of heat transfer, hardening or loss of strength. These parameters cannot be tracked visually. For the "range of [additional] materials," scoped in this program by the applicant, (stainless steel aluminum, copper, elastomers, PVC, thermoplastics, fiber glass, and fiber reinforced plastics) aging effects cannot be determined visually. The additionally scoped materials may render the External Surfaces Monitoring Program as implemented, not consistent with the GALL Report. Traditionally, the staff has considered the inclusion of these additional materials an exception to the GALL AMP.

Request:
1. Justify why the added range of materials is not an exception to GALL. Provide the basis for:
    a. Concluding that your External Surfaces Monitoring Program requires "enhancement" to be consistent with the GALL AMP, when your enhancements are beyond the scope of the GALL AMP.
    b. Including in your "consistent with GALL" program items not subject to loss of material by corrosion when the GALL AMP specifies detection of visible rust and rust byproducts (or visible coating degradation).
2. What inspection attributes will be implemented (e.g., parameter monitored/inspected, acceptance criteria, etc.), so that all SSCs identified in your LRA as covered by this AMP will be effectively monitored and managed for the aging effects cited?
3. As described in the scope for the program, describe how monitoring of degradation of paint and coatings will be performed.
4. For elastomers, identify the methods and procedures that will be used to identify loss of strength or hardening without physical manipulation of the elastomer.
5. For the following non-metallic materials identify methods or procedures to track wastage, loss of strength, and loss of material:
    a. PVC
    b. Thermoplastics
    c. Fiber Glass
    d. Fiber Reinforced Plastics

6. For the following metallic materials identify methods or procedures to track corrosion, oxidation, rust, and any other relevant aging effect:
    a. Copper
    b. Stainless Steel
    c. Aluminum
    d. How will reduction of heat transfer be monitored using this AMP?

RAI B.2.25-1

Background:
GALL AMP XI.S1, "ASME SECTION XI, Subsection IWE," recommends implementation of the ISI requirements of Subsection IWE, in accordance with 10 CFR 50.55a, and considers this a necessary element of aging management for concrete containments through the period of extended operation. This includes liner plate corrosion concerns described in the NRC generic communications.

Issue:
Page 43 of the CR-3 calculation L08-0616, "Description of the ASME Section XI, Subsection IWE Program," describes deterioration of the moisture barrier at reactor building 95 foot elevation. Liner plate thickness at this elevation was measured in 1997. Thickness of the liner in one area at the interface liner and at the moisture barrier location was measured by ultrasonic methods. The measured thickness was 0.307 inch as compared to nominal measured thickness of 0.390 inch (21% reduction in thickness).

Request:
Has the degraded area that was subjected to accelerated corrosion been UT examined in the successive outages since 1997 as recommended in IWE-1241 and Table IWE-2500-1 for augmented inspection? This information is required to determine if the effects of aging of the liner plate can be adequately managed during the period of extended operation as required by 10 CFR 54.21(c)(3).

RAI B.2.25-2

Background:
GALL AMP XI.S1, "ASME SECTION XI, Subsection IWE," recommends inspection of the containment liner plate in accordance with ASME Subsection IWE, Subarticle IWE-2411 and Table IWE-2411-1.

Issue:
According to Page 43 of the CR-3 calculation L08-0616, "Description of the ASME Section XI," the moisture barrier at containment base slab was reinstalled in 1997. However, the moisture barrier separation from liner plate and degradation was again documented starting in 2003. In 2007, the moisture barrier was found to be damaged at 12 locations with lengths of up to 36 inches. The damaged moisture barrier provides a path for water penetration at and below the floor level, and can affect the leak tightness of the containment during the period of extended operation.

Request:
Discuss any additional investigation and testing that are planned in addition to the visual examination of the moisture barrier during the 2009 refueling outage to determine the extent of liner plate corrosion at the moisture barrier and wall and floor liner plate below the moisture barrier. This information is required to determine if the effects of aging of the liner plate can be adequately managed during the period of extended operation as required by 10 CFR 54.21 (c)(3).

RAI B.2.25-3

Background:
GALL AMP XI.SI, "ASME Section XI, Subsection IWE," recommends inspection of the containment liner plate in accordance with ASME Subsection IWE, Subarticle IWE-2411 and Table IWE-2411-1.

Issue:
Action request AR 00257242 identified bulging in the liner plate at numerous locations. Additional investigation indicated hollow sounds at the bulge locations. This indicates separation of the liner plate from the containment concrete. In addition, AR 00257242 documented numerous failures in the coating for the liner plate.

Request:
1. Has any testing been performed to determine the gap between the liner plate and concrete? Provide details of any analysis performed to determine whether the separation of the liner is acceptable during all design basis loading conditions during the period of extended operation.
2. Is there a separate aging management program to monitor the containment liner plate coating degradation during the period of extended operation? If not what is the basis for evaluating the damage to the coating.

RAI B.2.26-1

Background:
GALL AMP XI.S2, "ASME Section XI, Subsection IWL," states that NRC IN 99-10 described occurrences of degradation in prestressing systems, and recommends that the applicant to consider the degradation in prestressing systems.

Issue:
The Operating experience section of LRA Section B.2.26 states that IN 99-10 was reviewed for applicability to CR-3. It was determined that the procedure used to control the tendon surveillance addressed the issues in the IN 99-10. The data for the CR-3 tendon history was reviewed using regression analysis, and the results did not vary appreciably from trending the group averages. However, LRA Section B.2.26 does not address the issue of high relaxation of prestressing steel wires at high operating temperature inside the containment. In addition, Calculation No. S07-033, Revision 0, Dated October 10, 2007, uses a loss in prestress due to relaxation to be only 2.95 percent at the end of 40 years. IN 99-10 reported a loss of prestress of 15.5 to 20 percent over a 40 year period at an average temperature of 90°F.

Request:
Explain how the loss of prestress of 2.95 percent due to relaxation of steel was determined and whether it is based on any test data.

Provide details of the informal review performed which determined that trending analysis for group averages is an acceptable method instead of the individual tendon lift-off forces linear regression analysis as recommended in IN 99-10.

RAI B.2.26-2

Background:
GALL AMP XI.S2, "ASME Section XI, Subsection IWL" states that trending and monitoring of prestressing forces in tendons for prestressed containments be in accordance with 10 CFR 50.55a(b)(2)(viii). In addition, 10 CFR 55.55a and ASME Subsection IWL also require that prestressing forces in all inspection sample tendons be measured by lift-off tests and compared with acceptance standards based on predicted force.

Issue:
During the several prestressing tendon surveillance inspections over the last 20 years, the lift-off forces in the hoop prestressing tendons have been consistently found to be lower than the 95 percent of predicted values. After the last (eighth) tendon surveillance in 2007, NCR 251318 disposition require follow up action to investigate the basis for the acceptance curves for tendons and determine why this criteria is typically more stringent than the life of plant curves.

Request:
What is the status of the investigation for the discrepancy in the actual lift-off and predicted forces for the prestressing tendons since it may affect the structural integrity of CR-3 containment during the period of extended operation?

RAI B.2.26-3

Background:
GALL AMP XI.S2, "ASME Section XI, Subsection IWL" references the American Concrete Institute (ACI) 201.1R-77 for identification of concrete degradation.

Issue:
Page 20-21, Section 6-3 of the CR-3 calculation L08-0617 states that ACI 201.1R-69 and R-92 were used in the development of the conditions indicative of degradation of IWL components, and use of the different editions of the ACI code is consistent with GALL.

Request:
Provide justification that use of ACI 201.1R-69 and R-92 editions are consistent with GALL without any exception.

RAI B.2.26-4

Background:
GALL report AMP X1.S2, “ASME Section XI, Subsection IWL” recommends that selected areas, such as those that indicate suspect conditions and areas surrounding tendon anchorages, receive a more rigorous VT-1 or VT-1C examination.

Issue:
It is not clear from Page 17-18 of the CR-3 calculation L08-0617 whether CR-3 inspects selected concrete surfaces that indicate suspect conditions and areas surrounding tendon anchorages by performing VT-1/ VT-1C examination or follow ASME Subsection IWL requirements.

Request:
Is the CR-3 inspection of selected areas of concrete that indicate suspect conditions and areas surrounding tendon anchorages consistent with GALL AMP X1.S2?

RAI B.2.28-1

Background:
Page 21 of the Crystal River Aging Management Program for 10 CFR 50, Appendix J (Calculation L08-0615), states that a technical specification change may be generated to take credit for IWE and IWL examinations.

Issue:
Appendix J of 10 CFR 50 requires a general inspection of the external and exterior surfaces to be performed prior to Type A test.

Request:
Explain how IWE examination performed during a period of 10 years can be credited for general inspection required to be performed prior to Type A Test.

RAI B.2.28-2

Background:
Appendix J, 10 CFR 50 specify halide leak-detection method or rate of pressure loss method as acceptable methods for performing Type B and C tests.

Issue:
CR-3 Type B and C tests are performed using make-up-flow method. In addition pages 19, 35, and 37 of calculation L08-0615 states that make-up-flow method as the NRC preferred method.

Request:
Provide justification for using make-up-flow method and documentation that indicate that make-up-flow as the NRC preferred method. Use of an unqualified method for leakage testing may affect the leakage rate results which may be used to predict containment performance during period of extended operation.

RAI B.2.28-03

Background:
10 CFR 50, Appendix J requires periodic performance of Type A test for detecting degradation of containment boundary.

Issue:
The table on Pages 22 and 23 of the CR-3 10 CFR 50 Appendix J Program (calculation L08-0615) provides results of the Type A test (ILRT) tests over the life of the plant. According to this Table, the containment leakage rate during 2005 test was two times of that recorded during the previous test performed during 1991.

Request:
Has the root cause for the 100 percent increase in the leakage rate between two successive tests been determined since it can affect the structural integrity of the containment to resist the design basis accident pressure load during the period of extended operation?

RAI B.2.30-1

Background:
According to operating experience described in the LRA section B.2.30 the Structures Monitoring Program is an existing program and currently the frequency of inspection is 10 years.

Issue:
The LRA states that the Structures Monitoring program is consistent with GALL AMP XI.S6. Also the program basis document states that the inspection criteria provided within the structures monitoring program are primarily taken from ACI 349.3R-96 which is in consistence with the GALL Report recommendation. Ten years inspection frequency for all structures and components is not in conformance with ACI 349.3R-96. ACI 349.3R-96, Chapter 6 recommends that the selected inspection frequency should provide assurance that any age-related degradation is detected at an early stage and that appropriate mitigative actions can be implemented. In addition, ACI 349.3R-96 also specifies a five-year (two per ISI interval) inspection for structures exposed to natural environment structures inside primary containment, continuous fluid exposed structures, and structures retaining fluid and pressure.

Request:
Provide justification for the inspection interval of 10 years and deviating from the ACI 349.3R-96 recommendations.

RAI B.2.30-2

Background:
LRA Section B.2.30 “Structures Monitoring Program” is an existing program that monitors aging of plant structures, including water controlled structures. This program corresponds to NUREG-1801, XI.S6, “Structures Monitoring Program,” and XI.S7, “Water Controlled Structures.” LRA and onsite basis document credits the applicant’s Structures Monitoring Program for “Inspection of Water-Control Structures Associated with Nuclear Power Plants.” The GALL Report allows this; however, the GALL Report requires that the details pertaining to water-control structures included in AMP B.2.30 are to incorporate the attributes described in GALL AMP XI.S7.

Issue:
A review of LRA Section B.2.30 indicates that the applicant has compared the attributes in this program to GALL AMP XI.S6 only. LRA B.2.30 does not include any comparison with the attributes of the program for water controlled structures (GALL AMP XI.S7) to establish consistency with the GALL Report.

Request:
Provide a comparison of attributes of LRA B2.30 to GALL AMP XI.S7 so as to determine if the B.2.30 program is consistent with GALL AMPs XI.S6 and XI.S7.

RAI B.2.30-3

Background:
The LRA Section B.2.30 states that the Structures Monitoring Program will be enhanced to monitor ground water chemistry including consideration for potential seasonal variation.

Issue:
The LRA has an enhancement for the periodic groundwater chemistry monitoring including the seasonal variation. However, the frequency of the groundwater chemistry monitoring is not specifically stated.

Request:
1. Describe past and present groundwater monitoring activities at CR-3.
2. What is the current ground water monitoring frequency and what will be the frequency of groundwater monitoring under the extended period of operation?
3. Provide the location(s) where test samples were/are taken relative to the safety-related and important-to-safety embedded concrete foundations.
4. Indicate seasonal variations.
5. Explain the technical basis and acceptance criteria.

RAI B.2.30-4

Background:
During the LRA audit, a plant walkdown was performed. Various concrete degradation mechanisms were observed on the walls of the Tendon Access Gallery at the 75 foot elevation. The noted deficiencies/aging effects include cracking, leaching, blistering, and voids. Water on the floor at several places was also noted.

Issue:
Various aging effects were observed. The source of water is still unknown. According to the engineering inspection report in the program basis document, the condition is acceptable and no corrective action is required.

The applicant's structures monitoring program, AMP B.2.30, is an existing program which was developed for the implementation of structures monitoring under 10 CFR 5.65, and is consistent with GALL AMP XI.56. The same program will be used for the extended period of operation. To meet the GALL Report recommendations the program element "Acceptance Criteria" is selected to ensure that the need for corrective actions will be identified before loss of intended function.

Request:
Provide an explanation of how the effect of aging will be adequately managed so that the intended function of protecting the tendon anchorage hardware against corrosion will be maintained consistent with the current licensing basis (CLB) for the period of extended operation without taking any corrective action.

RAI B.2.30-5

Background:
IN 2004–05 identified leakage of spent fuel pools at several existing nuclear power plants. The leak chase channels and associated piping were blocked by cementitious materials due to interaction with borated water.

Issue:
During a site walkdown of the spent fuel pool area on July 15, 2009, the staff observed that there was no leakage thru the leak chase channel piping located below the pool even though the valves on these piping were open. In addition, one of the pipe ends appeared to be blocked with cementitious material and boric acid crystals. Blockage of the leak chase channels can potentially cause leakage of the borated water from the spent fuel through the floor and walls of the spent fuel pool.

Request:
Provide a summary of the daily records of the leakage data collected at CR-3 spent fuel leak chase channel piping. The specific information required is as follows:

- When did the initial leakage of the leak chase piping stop?
- What action was taken to clean the leak chase piping?

This information is required to determine if there has been any degradation of the spent fuel pool concrete, liner, and rebar and how it will affect the integrity of the pool during period of extended operation.

RAI B.2.31-1

Background:
In LRA Section B.2.31, the applicant states that this is a new program and will be consistent with GALL AMP XI.E1, Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program. Under the Operating Experience (OE) Element, the applicant states that this program is a new program with no site specific OE history. However, under Element 4 of AMP B.2.31, the applicant states the CR-3 program utilizes plant OE to determine the plant areas to be inspected. It further states that based on this review of OE, the plant areas to be inspected become localized in nature, consisting of limited area (or subset) of a much larger plant area or zone. The corresponding GALL AMP XI.E1 program element states that a representative sample of accessible electrical cables and connections installed in adverse localized environment should be visually inspected for cable and connection jacket surface anomalies.

Issue:
It is not clear that Element 4 of the applicant's AMP is consistent with the corresponding element in the GALL AMP because the GALL Report recommends inspection of cables and connections installed in adverse localized environments while the applicant's AMP determines the areas to be inspected based on the plant OE.

Request:
Explain how the applicant AMP Element 4 is consistent with that in the GALL AMP, and how it will envelop electrical cables and connections in the scope of this aging management program.

RAI B.2.32-1

Background:
In the basis document L08-0641, "License Renewal Aging Management Program Description of Electrical and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits Program," under Detection of Aging Effects, the applicant states that as an alternate to the review of calibration or surveillance results, CR-3 will test the cable system used in the power range (PR) circuits of the excore monitoring system. In the corresponding GALL AMP element, states that in cases where a calibration or surveillance program does not include the cable system in the testing circuit, the applicant should perform cable system testing for detecting deterioration of the insulation system.

Issue:
GALL AMP XI.E2 recommends testing of cable system in nuclear instrumentation circuits disconnected during the calibration or surveillance procedures. The licensee's basis document, L08-0641, allows calibration of instrumentation system with the cable system disconnected during surveillance or calibration procedures.

Request:
Explain how the CR-3 AMP is consistent with the corresponding GALL AMP XI.E2 for detecting cable system deterioration.

RAI-B.2.35-1

Background:
For LRA AMP B.2.35, Program Element 3 (Parameters Monitored/Inspected) the applicant states, "Loss of continuity due to corrosion and oxidation will be managed by the Fuse Holder Program. Fatigue due to ohmic heating, thermal cycling, electrical transients, frequent manipulation, vibration, and chemical contamination are not applicable aging effects for CR-3 fuse holders located outside of active devices."

Issue:
GALL AMP XI.E3 states that this program element recommends monitoring thermal fatigue in the form of high resistance caused by ohmic heating, thermal cycling or electrical transients, mechanical fatigue caused by frequent removal/replacement of the fuse or vibration, chemical contamination, corrosion, and oxidation.

Request
Provide justification for eliminating the monitoring of thermal fatigue in the form of high resistance caused by ohmic heating, thermal cycling or electrical transients, mechanical fatigue caused by frequent removal/replacement of the fuse or vibration, and chemical contamination from LRA AMP B.2.35.

RAI B.3.1-1

Background:
LRA Section B.3.1 states that the CR-3 reactor coolant pressure boundary (RCPB) Fatigue Monitoring Program relies on monitoring and tracking the significant thermal and pressure transients for limiting RCPB components to prevent the fatigue design limit from being exceeded.

Issue:
The LRA provides no description or discussion regarding how CR-3 has been and will be monitoring the severity of pressure and thermal (P-T) activities during plant operations. It is essential that all thermal and pressure activities (transients) are bounded by the design specifications (including P-T excursion ranges and temperature rates) for an effective and valid aging management program.

Request:
1. Describe the procedure methods that CR-3 uses for tracking thermal transients.
2. Confirm that all monitored transient events are bounded by the design specifications.
3. Specify the time (years) over which actual transient monitoring and cycle tracking activities took place. If there have been periods for which transient events were not monitored since the initial plant startup, specify the affected time frame, and provide justification to demonstrate that the estimated cycles for this unmonitored period are conservative.

RAI B.3.1-2

Background:
LRA Section B.3.1, Operating Experience element, states that CR-3 performed a review of NRC generic communications (including Information Notices, Bulletins, Generic Letters, and draft generic communication), the Institute of Nuclear Power Operations OE database, and Licensee Event Reports, and identified no applicable OE items that relate to fatigue monitoring or to exceeding fatigue design limits.

Issue:
Page 15 of the AMP basis document L08-0604 indicates that CR-3's review of the NRC generic communications only covered the time since January 2005. It is unclear to the staff why the coverage was limited to such a short period. Industry OE reported in the earlier dates are just as valuable and important as the recently found ones and thus should be taken into consideration in developing effective aging managing programs also. For example, NRC Bulletin 88-08 affects all nuclear power reactors, and NRC Bulletin 88-11 affects all PWR plants. The issues and the unanticipated transients described in 88-08 and 88-11 are related to fatigue monitoring.

In addition, the onsite basis document states that "high cycle fatigue due to vibration is not germane to fatigue management program focused on tracking cycles and transients related to low cycle fatigue" and concluded that the OE on cracking/leaking in South Texas Project 2, Hope Creek, and St. Lucie 2 are not applicable to CR-3. It is unclear to the staff why the CR-3 fatigue AMP excludes high cycle fatigue effects.

Request:
1. Provide CR-3's review criteria that led to the conclusion as stated: "no applicable OE items that relate to fatigue monitoring or to exceeding fatigue design limits."
2. Describe the action that CR-3 continues to take in response to the NRC Bulletin 88-08.
3. Describe the action that CR-3 continues to take in response to the NRC Bulletin 88-11.
4. Provide basis that high cycle fatigue can be exempted from AMP when high cycle fatigue can also raise safety concerns, especially when high-cycle fatigue due to flow-induced vibrations is addressed in LRA Section 4.3.1.2 for the reactor vessel (RV) internal components.

RAI B.3.1-3

Background:
LRA Section B.3.1, Operating Experience element, states that CR-3 has reviewed the EPRI "Good Practice" documents related to fatigue and revealed that the CR-3 fatigue AMP is in accordance with the "Good Practice" recommendations.

Issue:
Additional information relating to the EPRI "Good Practice" is needed to enable the staff to perform its evaluation.

Request:
Provide a summary of the EPRI "Good Practice" (including EPRI report number) and demonstrate that the CR-3's fatigue monitoring program is consistent with the "Good Practice" recommendations.

RAI B.3.1-4

Background:
LRA Section B.3.1 states that the CR-3 Reactor Coolant Pressure Boundary (RCPB) Fatigue Monitoring Program will address the effects of the reactor coolant environment on component fatigue life at the sample locations identified in NUREG/CR-6260.

Issue:
It is unclear to the staff whether this feature has already been implemented in the CR-3 RCPB Fatigue Monitoring Program at the time of its license renewal application.

Request:
Confirm the status requested. If this feature has not yet been implemented, pursuant to the GALL Report requirements, it is necessary to make a license renewal commitment that the RCPB Fatigue Monitoring Program will be enhanced by monitoring the six component locations identified in NUREG/CR-6260 applicable to CR-3.

RAI B.3.1-5

Background:
CR-3 onsite basis document, L08-0604, states that the CR-3 RCPB Fatigue Monitoring Program Element 2, preventive action, is given an "alarm limit" feature that will be initiated when transient cycles in any category reaches 90% of the allowable value.

Issue:
It is unclear to the staff whether this feature has already been implemented in the CR-3 RCPB Fatigue Monitoring Program at the time of its license renewal application.

Request:
Confirm the status requested. If this feature has not yet been implemented, pursuant to the GALL Report requirements, it is necessary to make a license renewal commitment that the RCPB Fatigue Monitoring Program will be enhanced by providing the "alarm limit" capability.

RAI 4.3.1-1

Background:
In LRA Section 4.3.1, the applicant states that CR-3 performed an assessment of the number of nuclear steam supply system (NSSS) design transients that have occurred through December 2007 to determine the margin between the number of accrued cycles and the original 40-year design cycles. In addition, the applicant states that CR-3 has performed an assessment of the impact of the measurement uncertainty recapture 1.6 percent power uprate.

Issue:
Despite the statement indicated, throughout the whole LRA, no such data as the "accrued cycles" could be found. It is essential that the LRA provides the number of the accrued cycles to the date near its license renewal application submittal for all significant transients so as to facilitate future cycle prediction and management during the extended operation period. In addition, through the whole LRA, there is no discussion on the impact of the uprated power on the NSSS design transients.

Request:
1. Provide data of the accrued cycles for all transients that are managed and monitored under the CR-3 RCPB Fatigue Monitoring Program.
2. Describe how CR-3 assessed the measurement uncertainty recapture 1.6% power uprate and the impact of the power uprate on the NSSS design transients.

RAI 4.3.1-2

Background:
In LRA Section 4.3.1, the applicant states that based on 30 years of plant operating experience, for CR-3, there is considerable margin on the NSSS design transient cycles originally designed for 40 years and there is no need to increase the number of NSSS design transients for the period of extended operation. The applicant then made its 60-year projections on CUFs and transient cycles simply by multiplying a factor of 1.5 on both the 40-year CUFs and the design NSSS transient cycles.

Issue:
LRA Table 4.3-2, which contains the design fatigue usage factors (CUF, 40-year usage), shows that there are many locations that have CUF values greater than 0.67. Multiplying by 1.5 and these locations will have their CUF values greater than the limit of 1.0 during the extended operation period – inconsistent with the applicant's claim that there is considerable margin on design transient cycles and there is no need to increase the number of design transients for the period of extended operation.

Request:
1. Please provide the basis for the statement that there is considerable margin on the NSSS design transient cycles, and make necessary conforming corrections to the LRA.
2. Clarify whether the CUFs for the pressurizer nozzle, surge line hot leg nozzle and surge line elbows and piping as shown in Table 4.3-2 have included the insurge/outsurge and the stratification transients.

RAI 4.3.1.1-1

Background:
In LRA Section 4.3.1.1, the applicant disposes its TLAA for the RV components to 10 CFR 54.21(c)(1)(iii).

Issue:
It is unclear to the staff whether the disposition herein is meant to apply to all components of the RV, or only meant to apply to the only component that is projected to have its 60-year CUF exceed the limit. Uncertainty such as this one is seen in several other subsections under Section 4.3.

Request:
Clarify the uncertainty at the conclusion of Section 4.3.1.1 and other subsections that have the same uncertainty.

RAI 4.3.1.2-1

Background:
In LRA Section 4.3.1.2, five places show improbable number of cycles as basis to describe the endurance limit of fatigue life. Specifically, as appeared in the LRA, these are 1012, 106, 1012, 1013, and 1011 cycles.

Issue:
This is most likely formatting error. Endurance limit for metals usually are greater than 10^6 cycles. In addition, the LRA lacks the information about the material of the components considered for the FIV related high cycle fatigue analysis, and the maximum alternating stress (S_a) that was calculated in Report BAW-10051.

Request:
1. Correct the errors on the cycles accordingly.
2. Specify the material, temperature, and maximum alternating stress used in BAW-10051 for the fatigue analysis.
3. Provide the figure number and curve number of the ASME design S-N fatigue curve used for the endurance limit determination described in the LRA. Provide also the basis of choosing the fatigue curve used in your endurance limit calculation and the results.

RAI 4.3.1.3-1

Background:
In LRA Section 4.3.1.3, the applicant states that metal fatigue was considered in the design of the "Type C" control rod drive mechanism (CRDM) motor tube. However, the LRA also states that CUF of the CRDM motor tube was not calculated because the motor tube did not require analysis for cyclic operation in accordance with ASME Section III, paragraph N-415.1. In addition, the LRA also states that the NSSS design transients for CR-3 have not been increased for the period of extended operation. This same statement regarding the design cycles also appears in Section 4.3.1.4.

Issue:
It was uncertain to the staff whether CR-3 CUF for the CRDM motor tube was calculated or not. In addition, in LRA Section 4.3.1, the applicant states that the transient cycles are multiplied by a factor of 1.5 for the period of the extended operation.

Request:
1. Clarify whether the CUF analysis for the CRDM motor tube was completed.
2. Describe how ASME III Paragraph N-415.1 endorses exemption of fatigue usage calculation for the CRDM motor tube.
3. Provide basis that the transients for CR-3 are not increased for the period of extended operation. If the statement is false, correct accordingly for Sections 4.3.1.3 and 4.3.1.4.

RAI 4.3.1.4-1

Background:
In LRA Section 4.3.1.4, the applicant reviewed the transients and fatigue evaluations for various parts of the reactor coolant pump (RCP) and disposes the TLAA for the RCP locations in accordance with both 10 CFR 54.21(c)(1)(i) and 10 CFR 54.21(c)(1)(ii).

Issue:
The regulatory disposition statements should be part specific if not all parts of the analysis group consistently fall in the same disposition class.

Request:
Identify which part or locations of the RCP are managed in accordance with 10 CFR 54.21(c)(1)(i) and which are managed in accordance with 10 CFR 54.21(c)(1)(ii).

RAI 4.3.1.6-1

Background:
LRA Section 4.3.1.6 presents the fatigue TLAA for the pressurizer and shows the design fatigue values (40-year CUF) in Table 4.3-2.

Issue:
The LRA made no mention with regard to CR-3's position on stratification and insurge/outsurge events for pressurizer surge lines and response NRC Bulletin 88-11 that requires all PWR plants to include these thermal events in the fatigue evaluations to ensure ASME Code compliance.

Request:
1. Confirm whether the fatigue evaluations for the pressurizer surge nozzle including lower head region, surge line piping, and surge line hot leg nozzle have taken stratification and insurge/outsurge events into account.
2. Discuss how CR-3 reconstructed the heatup and cooldown cycles that occurred prior to December 20, 1988 (the date of issuance for Bulletin 88-11) for the pressurizer surge line stratification and insurge/outsurge events before the dates of issuance of NRC Bulletins 88-08 & 88-11.

RAI 4.3.2.1-1

Background:
LRA Section 4.3.2.1 shows the fatigue TLAA for the USAS B31.1.0 Class 1 piping. The section was divided into two parts. For the first part, it states that "Since the transient set (and associated cycles) in the Reactor Coolant System (RCS) Functional Specification is being

maintained, the analytical basis for these components remains unchanged." It then concluded that the TLAA for these components remain valid for the period of extended operation. Additionally, LRA Section 4.3.2.2 shows the fatigue TLAA for the USAS B31.1.0 Non-Class 1 piping. The applicant concluded that the TLAA for the portion of the Non-Class 1 piping components where the cycles are unrelated to the heatups and cooldowns can be projected to the end of the period of the extended operation.

Issue:
Adding the cycles from all transients shown in Table 4.3-1, the staff obtained 4957. To avoid double counting, the staff excluded 110 cycles from Transient 8d because this transient has been accounted in Transients 11, 15, 17A and 17B. In addition, the staff treated the heatup-cooldown as single events and so 240-cycle was counted only once. Multiplying 4957 by 1.5 and one obtains 7436 cycles for 60 years, which exceeds the 7000 cycles limit. However, the applicant did not make a required appropriate reduction to the allowable stress range. Similar situation occurs to a portion of the Non-Class 1 piping (LRA Section 4.3.2.2), namely, the portion where the cycles are unrelated to the heatup and cooldown transients. The 60-year cycles for these Non-Class 1 components also exceed the 7000 cycle limit, even though 240 cycles from the paired heatup-cooldown transients have been subtracted, but the applicant did not make a required appropriate reduction to the allowable stress range.

Request:
1. Please provide justification that the TLAA for the Class 1 piping components remain valid for the period of extended operation, 10 CFR 54.21(c)(1)(i).
2. Please provide justification that the TLAA for the portion of the Non-Class 1 piping components where the cycles are unrelated to the heatups and cooldowns can be projected to the end of the period of the extended operation, 10 CFR 54.21(c)(1)(ii).

RAI 4.3.3-1

Background:

LRA Section 4.3.3 discusses the environmentally assisted fatigue evaluation. On Page 4.3-12, it states that "Evaluations at all locations are based on application of environmental penalty factors to the ASME 40-year CUF values".

Issue:

Based on the CUF projection method described at the beginning of LRA Section 4.3, the projected 60-year CUF is 1.5 times the 40-year CUF value for each location. Therefore, environmentally assisted fatigue calculated based on 40-year CUF value as stated in LRA page 4.3-12 would be not conservative.

In addition, it is unclear to the staff what basis would lead to the numerical value 0.026, as shown in LRA page 4.3-12, for the transformed oxygen for stainless steels. Also, the CUF for the decay heat injection Class 1 piping stainless steel Tee under air environment is not included in LRA Table 4.3-2 but the environmentally adjusted CUF for this location is shown in Table 4.3-3. It is unclear to the staff how the CUF for this location was calculated.

Request:

1. Provide the basis that 40-year CUF instead of 60-year CUF can be used as the basis for calculating the environmentally adjusted CUF.
2. Confirm that the transformed oxygen for the stainless steel is 0.026 when the dissolved oxygen (DO) level is below the threshold value of 0.05 ppm, or provide a corrected value.
3. Provide the following input data used for the fatigue analysis for Decay Heat Injection Piping: Temperature; Transient set; Baseline CUF value being multiplied by the F_{en} factor.

RAI 4.3.3-2

Background:
LRA Table 4.3-3 shows the results of the environmentally assisted fatigue usage evaluations.

Issue:
Clarifications are necessary to enable the staff to complete its review. For example, on the 5th row, "Surge line piping up to but not including weld piping next to weld overlays (SS)" the F_{en} value is shown as a range of values instead of a single value for the location. In addition, several other issues concerning the environmental fatigue evaluation require clarification, as identified in the staff Request 2 through 7. In particular, it is unclear to the staff what the intention and basis are for the second sentence of the third paragraph on LRA Page 4.3-12, which reads, "The environmental correction factor F_{en} from NUREG/CR-5704 was used to determine the number of allowable cycles for each load pair".

Request:

1. Specify the F_{en} value used for the component described on the 5th row of the table.
2. Provide the source document that specified the extraneous 30,000 power loading and unloading transients in the CR-3 design basis as indicated in Note 1 under the LRA Table 4.3-3.
3. Specify the design cycles of the loading-unloading transients. Explain why there are 48000 cycles and 2600 cycles shows up in Note 2 under the LRA Table 4.3-3. Provide the basis for cycle reductions for this particular transient from 48000 to 2600.
4. Describe the revisions made to Transient 22, as indicated in Note 2 under LRA Table 4.3-3. Also, describe the impact of fatigue results due to such modifications.
5. Describe the role that the cited reference NUREG/CR-6717 plays in the CR-3 environmental fatigue analysis when, as stated in LRA Section 4.3.3, you used NUREG/CR-6583, NUREG/CR-5704, and an open literature for calculating F_{en} for components made of carbon/low-alloy steels, stainless steels, and nickel alloys, respectively. Also, please identify the open literature mentioned here.
6. Summarize the integration method, and provide references, stated in the third paragraph on LRA Page 4.3-12, which you used for F_{en} determination.
7. Clarify the last part in the Issue segment above. What are the allowable cycles? Explain why F_{en} can be used as basis to determine the number of the so-called allowable cycles.

RAI 4.3.3-3

Background:
LRA Section 4.3.3 describes the environmental fatigue evaluation and the results are presented in LRA Table 4.3-3, including the F_{en} values determined for each component or location evaluated.

Issue:
LRA Table 4.3-3 shows a F_{en} value of 2.45 being used for all of the locations that use low alloy steels. It is known that F_{en} depends on material, strain rates, temperature and the dissolved oxygen (DO) concentration of the reactor water. For low alloy steel and carbon steels to maintain at this particular F_{en} value, 2.45, it requires that the DO is maintained at or below the threshold level of 0.05 ppm.

Request:
1. Summarize CR-3's experience in control of DO level in the reactor water since the plant startup. Describe all water chemistry programs CR-3 has used, including procedures and requirements used for managing DO concentration as well as the inception date of each water chemistry program.
2. Provide a historic summary of the DO level since plant startup. Estimate the fraction of time of the CR-3 operating history thus far that the DO level exceeded 0.05 ppm.
3. Describe how reactor water samples were taken, including the sampling locations. If samples were taken from a single location, justify that the DO data sampled are applicable to all NUREG/CR-6260 locations in CR-3 for the F_{en} calculations.

RAI 4.3.3-4

Background:
The results of weld overlay application for the surge line hot leg nozzle were unacceptable due to presence of indications (flaws) in the weld deposit and the overlay weld was removed.

The results of the weld overlay application for the surge line pressurizer nozzle were acceptable.

Issue:
The unacceptable weld overlay for the surge line hot leg nozzle raises concerns on the validity of the environmentally adjusted CUF values for the 5th and 6th items shown in LRA Table 4.3-3 "Surge line piping up to but not including weld piping next to weld overlays (SS); and "Surge line hot leg nozzle and stainless steel piping adjacent to weld overlay (SS) respectively since the CUFs of these two locations will be affected by the: (1) application of the weld overlay; (2) removal of the weld overlay; and, (3) reapplication of weld overlay, if this occurs. These above activities cause the stress at the weld overlay and surrounding areas to deviate from the stress state defined in the CLB.

During the audit, the applicant indicated that reapplying the weld overlay for the surge line hot leg nozzle is scheduled for the upcoming refueling outage. The staff noted that a written license renewal commitment for reapplying this weld overlay was not included in the LRA

Furthermore, the CUF validity concern extends to the weld overlay on the surge line pressurizer nozzle since the stress state for the 7th item of LRA Table 4.3-3 will deviate from the stress state defined in the CLB because of the existence of the weld overlay.

Request:

1. Describe how the CUFs shown in LRA Table 4.3-3 for the 5th, 6th, and 7th items (as described above) were determined.
2. Reassess the CUF for LRA Table 4.3-3, Items 5 and 6 when the weld overlay is reapplied.
3. Describe the transient set and cycles used for CUF calculations for the three components (locations) of concern. Confirm that the cycles used are the 60-year projected values.
4. Clarify whether or not a full structural weld overlay for the surge line hot leg nozzle will be reapplied.
5. Discuss the purpose of the full structural weld overlay for the pressurizer surge nozzle and the surge line hot leg nozzle.
6. Provide a discussion of any other structural changes made that could affect fatigue results but are not already discussed in the LRA.

RAI 3.3.2.2.1-1

Background:
LRA Sections 3.3.2.2.1 (for Auxiliary Systems) and 3.4.2.2.1 (for Steam and Power Conversion Systems), both state that the TLAAs on fatigue are addressed separately in Section 4.3. In addition, both of these LRA sections involve line items cited with Note J (identified in LRA Tables 3.3.2 Series and LRA Table 3.4.2 Series), which indicates that neither the component nor the material and environment combination for the components in these systems are evaluated in NUREG-1801.

Issue:
It is unclear to the staff whether LRA Section 4.3 has covered fatigue TLAA for the components under groups of Auxiliary Systems (AUX), and Steam and Power Conversion (SPC), as the applicant claimed. Specifically, LRA Table 3.3.2 series and Table 3.4.2 series identified that the following components are managed by means of TLAA evaluation:

> EFP-3 Diesel Engine Exhaust Expansion Joints and Silencers, standpipes, hydrants, and tanks; Deaerator, Expansion Joints, Feedwater Booster Pumps, tanks, Feedwater Heaters, Main Feedwater Pumps.

The staff was not able to locate any of these items in LRA Section 4.3 as the applicant stated. Note that the above list does not include piping, which is covered under LRA Section 4.3.2.2 for B31.1.0 Non-Class 1 piping.

Request:

1. Identify under which subsections of LRA Section 4.3 these components are covered.
2. Discuss the methods used for the TLAA analysis for these components.
   projected