May 9, 2003

LICENSEE: South Carolina Electric and Gas Company

FACILITY: V. C. Summer Nuclear Station

SUBJECT: SUMMARY OF THE MEETING BETWEEN THE U.S. NUCLEAR REGULATORY COMMISSION (NRC) STAFF AND SOUTH CAROLINA ELECTRIC AND GAS COMPANY REPRESENTATIVES TO DISCUSS THE V. C. SUMMER NUCLEAR STATION (VCSNS) LICENSE RENEWAL APPLICATION (LRA)

On February 26-27, 2003 and March 5, 2003, the NRC staff met with members of South Carolina Electric & Gas Company (SCE&G) in two separate public meetings. The purpose of the two meetings was to discuss the staff's draft request for additional information (RAI) in Sections 2.0, 3.1, 3.2, 3.3, 3.4, 4.0 and Appendix B to their LRA submittal. The meetings were useful in (i) combining duplicate questions, (ii) clarifying the intent of and expectations for a satisfactory response to the staff's draft RAIs, and (iii) omitting questions for which the requested information was already available in the LRA or other docketed correspondences. While the resolution of the draft RAIs was based on information already available in the docketed LRA material, clarifications and augmentation will be provided by the licensee via formal response to the RAIs. The safety RAIs were finally issued on March 28, 2003. This summary formalizes the meeting discussions.

The combined list of attendees for the two meetings is provided in Enclosure 1. The draft RAI questions that were circulated in the two-day meetings is provided in Enclosure 2.

Highlights of the two meetings are as follows:

i. These RAI questions were repetitive in that, the issues that needed resolution by different NRC staff were the same (e.g. chemistry program), though the specific systems involved were different (e.g. pressurizer and steam generator). These were combined in the final RAI for the following questions.

3.1.2.4.2	Boric Acid Corrosion and Surveillance
3.3.2.2.6	Loss of material due to Galvanic, General, pitting and Crevice corrosion
3.1.2.4.4	Neutron Noise Monitoring
3.1.2.4.6	Chemistry Program

ii. Draft RAI questions that were clarified, revised or added include the following RAI questions. As seen from the applicant's initial draft responses, there were instances where the applicant appeared to not fully understand the intent or the nature of the staff questions. In respect of these questions the staff provided appropriate clarifications during the meeting and also indicated what the staffs expectation of the responses.

2.3.5-1	Circulating Water System
3.3.3.4.7-1	Diesel Generator Services System

3.3.3.4.7-3	Diesel Generator Mufflers
3.3.2.4.14-2	Liquid Waste System
B.2.7-1	Small Bore Piping Inspection

iii. Draft RAI questions that were resolved and questions deleted as follows. With respect of these questions, the staff requested and the applicant readily pointed out in appropriate sections in the LRA, where the answers to the staff questions were already available. These references were, otherwise not readily discernable in the LRA. Where the staff was satisfied with the clarifications those questions were treated as resolved and deleted from the final issuance of the RAI.

2.3.3.5-1	Circulating Water System
2.3.3.10-3	Gaseous Waste Processing System
2.3.3.22	Spent Fuel Cooling System
2.3.4.7-12	Steam and Power Conversion System

There were no members of the public present during the meeting.

A draft of this meeting summary was provided to the applicant to allow them the opportunity to comment prior to the summary being issued.

/***RA***/

Ram Subbaratnam, Project Manager
License Renewal Section
License Renewal and Environmental Impacts Programs
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Docket No.: 50-395

Enclosures: As stated

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3.3.3.4.7-3	Diesel Generator Mufflers
3.3.2.4.14-2	Liquid Waste System
B.2.7-1	Small Bore Piping Inspection

ii Draft RAI questions that were resolved and question deleted as follows. With respect of these questions, the staff requested and the applicant readily pointed out in appropriate sections in the LRA, where the answers to the staff questions were already available. These references were, otherwise not readily discernable in the LRA. Where the staff was satisfied with the clarifications those questions were treated as resolved and deleted from the final issuance of the RAI.

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SOUTH CAROLINA ELECTRIC & GAS COMPANY
FEBRUARY 26-27, AND MARCH 5, 2003

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* Participated via telephone for the meeting on February 26, 2003

Enclosure 1

Summer LRA
Draft RAIs on AMPs
EMEB

<u>AMP B.1.9 Service Water System Reliability and In-Service Testing Program</u>

No RAIs

(RAI to be deleted. We will assume that the statement, "consistent with GALL," implies that it covers the recommendations in GL 89-13. This will be verified during the inspection, and the SER will be changed accordingly. This will be passed to the PM as an inspection item.)

<u>AMP B.1.19 Material Handling System Inspection Program</u>

RAI B.1.19-1

The staff's position, as described in GALL Vol. 2 item VII.B.2-a, is that wear on crane rails falls within the scope of license renewal, even though it is caused by active components. The crane rails are passive, long-lived components, and wear is an applicable aging effect. Please clarify whether the Material Handling System Inspection Program (B.1.19-1) is capable of detecting wear on the crane rails, and provide the appropriate updates to the 10 elements of this program.

RAI B.1.19-2

The LRA is not clear which cranes are covered by this AMP. The only reference to this AMP is from AMR Table 3.3-1, Item 15; however, there are no LRA Section 2 tables that refer to this AMR item. Please clarify the AMR of the cranes, and clarify which cranes use the Material Handling System Inspection Program.

<u>AMP B.1.25 Preventive Maintenance Activities-Terry Turbine</u>

RAI B.1.25-1

The Preventive Maintenance Activities-Terry Turbine description states in LRA Section B.1.25, under element 5 ("Monitoring and Trending"), that "routine periodic visual inspections are conducted…in order to detect age-related degradation and to initiate corrective actions as necessary." Please specify the frequency of these periodic inspections or how the inspection frequency is determined.

3.6-1 The applicant states that the Non-EQ Insulated Cables and Connections Inspection Program will be consistent with XI.E1, Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements, as identified in NUREG01801 prior to the period of extended operation. However, the program discussed in 3.6-1 does not agree with NUREG 1801. Explain (by comparing each element of the XI.E1 and the applicant's AMP) how the applicant's AMP is consistent with the GALL program XI.E1.

3.6-2 Exposure of electrical cables to localized environments caused by heat, radiation, or moisture can result in reduced insulation resistance (IR). Reduced IR causes an increase in leakage currents between conductors and from individual conductors to ground. A reduction in IR is a concern for circuits with sensitive, low-level signals such as radiation monitoring and nuclear instrumentation since it may contribute to inaccuracies in the instrument loop. Visual inspection may not be sufficient to detect aging degradation from heat, radiation, or moisture in the instrumentation circuits with sensitive, low-level signals. Because low-level signal instrumentation circuits may operate with signals that are normally in the pico-amp or less, they can be affected by extremely low levels of leakage current. These low levels of leakage current may affect instrument loop accuracy before the adverse localized changes are visually detectable. Routine calibration tests performed as part of the plant surveillance test program can be used to identify the potential existence of this aging degradation. Provide a description of your aging management program that will be relied upon to detect this aging degradation in sensitive, low-level signal circuits.

3.6-3 Table 3.6-1, Item 4 of the LRA indicates that the aging management review for medium voltage cables exposed to moisture and voltage stressors concluded that aging management at VCSNS is not required. No instances of power cable failure at VCSNS due to moisture intrusion were found.

Most electrical cables in nuclear power plants are located in dry environments. However, some cables may be exposed to condensation and wetting in inaccessible locations, such as conduits, cable trenches, cable troughs, duct banks, underground vaults or direct buried installations. When an energized medium-voltage cable is exposed to wet conditions for which it is not designed, water treeing or a decrease in the dielectric strength of the conductor insulation can occur. This can potentially lead to electrical failure. The growth and propagation of water trees is somewhat unpredictable. Provide a description of your aging management program that will be relied upon to provide reasonable assurance that the intended function of inaccessible medium-voltage cables that are not subject to environment qualification requirements of 10 CFR 50.49 and are exposure to moisture while energized will be maintained consistent with the current license basis through the period of extended operation.

3.6-4 In Table 3.6-2, of the LRA , the applicant states that aging effects of non-EQ electrical penetration assemblies include embrittlement, cracking, melting, discoloration, swelling, or loss dielectric strength leading to reduced insulation resistance, electrical failure caused by thermal/thermoxidative degradation of organics, radiolysis and photolysis (ultraviolet sensitive materials only) of organic, radiation-induced oxidation, moisture intrusion. However, the applicant states that for the ambient environmental conditions at VCSNS, no aging effects have been identified that could cause a loss of function and no aging management is required.

In most areas within a nuclear power plant, the actual ambient environments are less severe than the nominal plant environment. However, in a limited number of localized areas, the actual environments may be more severe than the nominal plant environment. Insulation materials used in Non-EQ electrical penetration assemblies may degrade more rapidly than expected in these adverse localized environments. The purposed of the aging management program is to provide reasonable assurance that the intended functions of electrical penetration assemblies exposed to adverse localized environment caused by radiation or heat will be maintained to be consistent with the current licensing basis through the period of extended operation. For Non-EQ electrical penetration within the scope of license renewal exposed to adverse localized environments, provide a description of an aging management program for electrical penetration insulation exposed to an adverse localized environmental caused by heat, radiation, or moisture.

3.6-5 In a letter dated March 4, 2003, the NRC forwarded to the Nuclear Energy Institute (NEI) and Union of Concerned Scientists, an interim staff guidance (ISG) on the identification and treatment of electrical fuse holders. The staff position indicated that fuse holders should be scoped, screened, and included in the aging management review (AMR) in the same manner as terminal blocks and other types of electrical connections that are currently being treated in the process. This position only applies to fuse holders that are not part of a larger assembly such as switchgear, power supplies, power inverters, battery chargers, circuit boards, etc. Fuse holders in these types of active components would be considered to be a piece parts of the larger assembly and not subject to an AMR.

Operating experience as discussed in NUREG-1760 (Aging Assessment of Safety-Related Fuses Used in Low- and Medium-Voltage Applications in Nuclear Power Plants) identified that aging stressors such as vibration, thermal cycling, electrical transients, mechanical stress, fatigue, corrosion, chemical contamination, or oxidation of the connections surfaces can result in fuse holder failure. On this basis, fuse holders (including both the insulation material and the metallic clamps) are subject to both an AMR and AMP for license renewal. Typical plant effects observed from fuse holder failures due to aging have resulted in: challenges to safety systems, cable insulation failure due to over-temperature, failure of a containment spray pump to start, a reactor trip, etc. Therefore, managing age-related failures of fuse holders would have a positive effect on the safety performance of a plant. Implement the fuse holder ISG.

3.6-6 Explain in details why connection surface oxidation of high voltage electrical switchyard bus are not considered significant aging mechanism at the Virgil C. Summer Nuclear Station (VCSNS).

3.6-7 The most prevalent mechanism contributing to loss of high voltage transmission conductor strength is corrosion which includes corrosion of steel core and aluminum strand pitting. Explain in details why no aging effects related to conductor corrosion have been identified that would cause a loss of function for the extended period of operation. Also, explain why no significant aging effects related to wind loading vibration or sway on high voltage connections has been identified at VCSNS.

3.6-8 Various airborne materials such as dust, salt and industrial effluents can contaminate insulator surfaces. A large buildup of contamination enables the conductor voltage to track along the surface more easily and can lead to insulator flashover. Surface contamination can be problem in areas where there greater concentration of airborne particles such as near facilities that discharge soot or near the sea coast where salt spray is prevalent. Cracks have been known to occur with insulators when the cement that binds the parts together expands enough to crack the porcelain. Mechanical wear is another aging effect for strain and suspension insulators in that they are subject to movement. Movement of insulators can be caused by wind blowing the supported transmission conductor, causing it to swing from side to side. If this swinging is frequent enough, it could cause wear in the metal contact points of the insulator string and between an insulator and the supporting hardware. Provide a detail assessment of surface contamination, cracking, and loss of material due to wear for high voltage insulators and explain why these potential aging effects are not significant for VCSNS.

Request for Additional Information (RAI)
V. C. Summer Nuclear Station License Renewal Application
Section 3.5, “Aging Management of Containments, Structures, and Component Supports”
and
Section 4.5, “Concrete Containment (Reactor Building) Tendon Pre-Stress Analysis”

RAI 3.5-1
In Report TR00170-003, Revision 0, Attachment II: Aging Management Review Results for Structures and Structural Components, cable trays, conduit, electrical and instrument panels and enclosures are identified as component types within most of the buildings and structures. These components are identified as steel in an internal environment, except for the Electrical Substation and Transformer Area, where the environment is external. In all cases, no aging effect requiring aging management is identified. The staff believes that these components located in the reactor, auxiliary, intermediate, and fuel handling buildings are susceptible to boric acid corrosion and that these components located in an external environment are susceptible to environmental corrosion. Therefore, in both cases loss of material is an applicable aging effect requiring aging management. The applicant is requested to identify and describe the aging management programs which will manage loss of material for these components located in the reactor, auxiliary, intermediate, and fuel handling buildings, and in an external environment.

RAI 3.5-2
Many concrete component types in internal, external, and below-grade environments are identified in Report TR00170-003, Revision 0, Attachment II as having no aging effects requiring aging management. The specific component types are duct banks; equipment pads; flood curbs; foundations; hatches; missile shields; reinforced concrete-beams, columns, floor slabs, walls; roof slabs; sumps; caissons; piers; trenches; jet barriers; and manholes. The staff position is that all accessible concrete components that perform an intended function require aging management for loss of material, cracking, and change in material properties; and that inaccessible concrete components (i.e., below grade) also require aging management unless specific criteria defined in NUREG-1801 GALL Volume 2 are satisfied, to demonstrate a non-aggressive below-grade environment. Therefore, the applicant is requested to

(d) identify the aging management programs which will manage loss of material, cracking, and change in material properties for all concrete components in accessible areas;
(e) submit a quantitative assessment of the below-grade environment that demonstrates it is non-aggressive, in accordance with the specific criteria defined in GALL Volume 2, and describe the groundwater monitoring program that will be implemented to verify that the below-grade environment remains non-aggressive, including monitoring frequency and consideration of seasonal fluctuations;
(f) if the below-grade environment is aggressive, in accordance with the specific criteria defined in GALL Volume 2, describe in detail the plant-specific aging management programs for inaccessible concrete components.

RAI 3.5-3
Report TR00170-003, Revision 0, Attachment II does not list containment hatch O-rings or fire/flood door seals as separate components. Therefore, there is no documented aging management review. Since these components are passive and are typically replaced only upon

identification of a degraded condition, they require an aging management review. Therefore, the applicant is requested to submit its aging management review for these components, including a description of the aging management programs that will be relied upon to ensure there is no loss of intended function during the period of extended operation.

RAI 3.5-4

In Report TR00170-003, Revision 0, Attachment II, three (3) steel components in an concrete environment are listed. These are anchorage, anchorage/embedments (exposed surfaces) and embedments. All are identified as having no aging effects requiring aging management. The condition of the concrete surrounding anchorage and embedments may affect their load capacity. GALL Volume 2, III.B, Item Numbers III.B1.1.4, III.B1.2.3, III.B2.2, III.B3.2, III.B4.3, and III.B5.2 specifically identify the need for aging management of the concrete surrounding expansion and grouted anchors, and grout pads for support base plates. Localized concrete degradation may have an insignificant effect on the overall structure integrity, but may have a very significant effect on the anchor capacity. The staff position is that all accessible concrete requires aging management; this includes monitoring the condition of concrete surrounding anchorages and embedments. Therefore, the applicant is requested to submit its detailed aging management review for anchorage and embedments, including a description of the aging management programs that will be relied upon to manage this aging effect.

RAI 3.5-5

In Report TR00170-003, Revision 0, Attachment II, for pipe supports located in the auxiliary building; control building; intermediate building; diesel generator building; fuel handling building; reactor building; and service water structures, loss of material is identified as the only aging effect requiring aging management. The ASME Section XI ISI Program - IWF is identified as one of the credited aging management programs, presumably for ASME Class piping supports. Attachment II indicates that this is a match with GALL. The staff notes that this is not a match with GALL, because GALL Volume 2, III.B, Item Numbers III.B1.1.3 and III.B1.2.2 also identify loss of mechanical function as an aging effect to be managed by IWF. Therefore, the applicant is requested to revise Attachment II and the license renewal application (as appropriate) to include loss of mechanical function as an applicable aging effect for ASME Class piping supports, and to credit IWF as the applicable aging management program. Alternatively, submit a detailed technical basis for excluding this aging effect and clearly identify this as a deviation from GALL.

RAI 3.5-6

In the "Aging Management Programs" column of the Report TR00170-003, Revision 0 Attachment II Tables, Technical Specification 3/4.9.10 is listed for the fuel transfer canal liner plate, spent fuel pool liner, and spent fuel storage rack in the fuel handling building; and Technical Specification 3/4.6.1 is listed for personnel airlock, escape hatch, and equipment hatch in the reactor building. The staff requests the applicant to describe the objective, scope, and implementation procedures of each technical specification, as it relates to aging management for license renewal.

RAI 3.5-7

In LRA Section 3.5.1.2, the applicant has identified that the foundation for the auxiliary building extends below the groundwater level and is supported on fill concrete down to competent bedrock. However, the applicant did not identify whether underdrain (de-watering) systems are utilized at V. C. Summer for the auxiliary building and/or any of the other buildings in the license renewal scope. In addition, no intended function(s) have been identified for the fill concrete used under several of the buildings included in the license renewal scope. Therefore, the staff requests the applicant to submit the following information related to underdrain systems and fill concrete:

(g) Identify whether underdrain (de-watering) systems are utilized at V. C. Summer.
(h) If utilized, describe the specific applications; describe current monitoring and/or maintenance activities that ensure proper functioning; discuss whether they perform an intended function; and, as appropriate, submit an aging management review, including identification of credited aging management program(s).
(i) Discuss whether fill concrete performs an intended function; and, as appropriate, submit an aging management review, including identification of credited aging management program(s).

RAI 3.5-8
In LRA Table 3.5-1, AMR item 19, the applicant credits the Chemistry Program (LRA Appendix B.1.4) for aging management of the stainless steel, spent fuel pool liner. The staff considers verification of the effectiveness of a chemistry control program to be an integral element of aging management. For the spent fuel pool, this is readily achieved by monitoring an existing plant-specific, spent fuel pool leak detection system or by monitoring the spent fuel pool water level for indications of leakage. Therefore, the staff requests the applicant to describe its plant-specific operating experience concerning leaks in the spent fuel pool, including a description of each occurrence, how it was detected, the determination of root cause, and how it was remedied.

RAI 3.5-9
LRA Table 3.5-2 is titled “Summary of Aging Management Programs for Station Containment, Other Structures and Component Supports That are Different From or Not Addressed in NUREG-1801 but are Relied on for License Renewal.” Ten (10) AMR items are listed in the table. For each AMR item, the following information is provided in the table: component type, material, environment, aging effect / mechanism, program activity, and discussion. The staff’s review of LRA table 3.5-2 identified the need for clarification and additional information relating to a number of the AMR items. For all except one (1) of these items, additional pertinent information has either been requested in other RAIs or was located in Attachment II to Report TR001700-003. The exception is LRA Table 3.5-2, AMR item 4: “Lubrite Plates (Class 1 Pipe Hanger Supports).” It is identified as a lubricant material in an internal environment. No aging effect / mechanism is identified, and consequently no aging management program is identified. In the “Discussion” column, the applicant provided a brief summary of its aging management review, which concluded that lubrite plates “are not susceptible to aging effects requiring management.” Aging management of lubrite plates for Class 1 piping supports is addressed in NUREG-1801, GALL Volume 2, III.B, Item No. III.B1.1.3. ASME Section XI, subsection IWF is identified as the applicable aging management program. Therefore, the applicant is requested to submit a detailed technical basis to support its conclusion that lubrite plates do not require aging management, or revise its aging management review for lubrite plates to be consistent with GALL.

RAI 3.5-10
LRA Table 3.5-1 is titled “Summary of Aging Management Programs for Station Containment, Other Structures and Component Supports Evaluated in NUREG-1801 That are Relied on for License Renewal.” Twenty-nine (29) AMR items are listed in the table. For each AMR item, the following information is provided in the table: component group, aging effect / mechanism, aging management program, further evaluation required, and discussion. This table is a reproduction of NUREG-1800 Table 3.5-1, with an added “Discussion” column. LRA Table 3.5-1 does not indicate that the applicant’s aging management reviews are consistent with GALL. In the “Discussion” column, the applicant refers to aging management programs that are “consistent with those reviewed and approved in NUREG-1801.” For most of the AMR items, the aging management review is not consistent with GALL. The staff’s review of LRA table 3.5-1 identified the need for clarification and additional information relating to many of the AMR items. For many of these items, additional pertinent information has either been requested in other RAIs or was located in Attachment II to Report TR001700-003. The applicant is requested to submit the following additional information or clarifications related to LRA Table 3.5-1:

(j) For AMR items 1 and 2, describe how the design basis for the flat plate containment penetration closures considered cyclic loading due to temperature/pressure transients. If a CLB fatigue analysis exists for the flat plate penetration closures, has it been updated for a 60-year operating life? How will cracking due to cyclic loading be managed for the period of extended operation?
(k) For AMR item 8, clarify the reference to three (3) aging management programs in the “Discussion” column, considering that the containment foundation is not subject to settlement.
(l) For AMR item 15, clarify the reference to three (3) aging management programs in the “Discussion” column, considering that freeze-thaw and reaction with aggregates are dispositioned as not requiring aging management for both accessible and inaccessible areas.
(m) For AMR item 16, explain the reference to two (2) aging management programs that are only applicable to the containment structure.
(n) For AMR item 24, explain the following statement in the “Discussion” column: “Note that the combinations of components, materials, and environments identified in NUREG-1801 for Group 8 (Steel Tanks) are not applicable to VCSNS; therefore, aging management is not required.” Do any steel tanks have stainless steel liners? If so, how are SCC and crevice corrosion managed?
(o) For AMR item 25, clarify which listed subcomponents are managed by each of the two (2) referenced aging management programs. Also identify which, if any, of the subcomponents do not require aging management, based on the plant-specific aging management review.
(p) For AMR item 28, explain why ASME Section XI, subsection IWF is not credited for aging management of the ASME Class supports, consistent with GALL. How are the two (2) referenced aging management programs implemented as a substitute for IWF?

RAI 3.5-11
LRA Section 3.5.1.1 indicates that the reactor building foundation mat bears on fill concrete that extends to competent rock. A retaining wall, extending approximately one-quarter of the way around the reactor building, protects the below grade portions of the reactor building wall from the subgrade. LRA Section 2.4.1 indicates that the retaining wall protects the below-grade

portions of the reactor building wall from the subgrade and groundwater. The groundwater at VCSNS has been identified as being mildly acidic but considered to be not aggressive in LRA Table 3.5-1. No quantitative information is provided in the LRA Section 3.5 to justify this conclusion. Therefore, the following information is requested:

(q) Identify the intended function(s) for the retaining wall, or provide the technical basis why it serves no intended function.
(r) Provide the aging management review for the retaining wall, if it serves any intended function.

RAI 3.5-12
AMR items 7 and 15 of LRA Table 3.5-1 indicate that only certain aging effects of concrete containment require aging management. As an example, for accessible exterior portions of the reactor building concrete containment, only change in material properties due to leaching requires aging management in accordance with the Containment ISI Program - IWE/IWL. For inaccessible areas, sufficient information (specific standards and quantitative data) was not provided to clearly demonstrate that containment concrete aging effects do not require aging management. Therefore, the following information is requested for containment concrete components:

(s) For all accessible containment concrete components, demonstrate that cracking, loss of material, and change in material properties will be managed in accordance with NUREG-1801, XI.S2, ASME XI, Subsection IWL.
(t) For all inaccessible containment concrete components, describe the plant-specific aging management programs which will manage aging unless it is clearly demonstrated that the non-significance conditions specifically described in the NUREG-1801 apply.

RAI 3.5-13
In Report TR00170-003, Rev. 0, Attachment II, many structural components are identified as not having any applicable aging effects and thereby no aging management programs are specified in the “Aging Management Programs” column. Most of these structural components are concrete, which the staff addresses in RAI 3.5-2. For several stainless steel components in the reactor building (refueling canal liner plate, sump screens, and sumps), a statement in the “Notes” column indicates that although no aging effects have been identified, the Maintenance Rule Structures Program inspects these components. Please explain the intent of this statement. Is the Maintenance Rule Structures Program being credited to manage aging of these components for license renewal?

RAI 3.5-14
AMR item 10 in LRA Table 3.5-1 addresses the aging effect of reduction in strength and modulus due to elevated temperature for concrete elements of containment. The discussion column of this item states that “The VCSNS containment concrete elements are not exposed to temperatures which exceed the thresholds for degradation; therefore, reduction of strength and modulus due to elevated temperatures are not aging effects requiring management. This statement does not seem to be consistent with the information presented in Report TR00170-003, Rev. 0, Table 6.1-1 and discussion on page 59 of 224. The table indicates that there is one region (above the reactor head but below the operating floor elevation 463') that has a maximum temperature of 157° F. Page 59 of the report also indicates that the control rod drive

mechanism (CRDM) is maintained at a temperature of less than or equal to 170° F. The report concludes that these temperatures are localized and do not exceed 200° F. The report follows with some additional discussion about elevated temperature concerns for three areas inside the reactor building. Some design modifications were made to rearrange air flow in the reactor building and tests were made in which the inspector identified no further problems. From this information it is not clear how many regions still have temperatures above 150° F and how the aging effects due to elevated temperatures above 150° F will be managed. Therefore, provide the following information:

(u) Explain the inconsistency between the statement made in LRA Table 3.5-1, AMR item 10 and the information in Report TR00170-003, Rev. 0, Table 6.1-1 (see above discussion).
(v) For all structures in the scope of license renewal, identify all regions that currently have temperatures in excess of 150° F.
(w) How will aging effects due to elevated temperatures above 150° F be managed during the period of extended operation?
(x) What is the meaning of the phrase "the inspector identified no further problems" following design modifications to rearrange air flow in the reactor building?

RAI 3.5-15

AMR item 12 in LRA Table 3.5-1 discusses loss of material due to corrosion in accessible and inaccessible areas of the containment liner. For inaccessible areas, the LRA concluded that corrosion in the embedded containment liner is not significant because the four conditions described in NUREG-1801 are applicable to VCSNS. The staff notes that the plant-specific operating experience does not necessarily support this conclusion. LRA Appendix B.1.12.1, states that rust was identified on the reactor building liner plate adjacent to the moisture barrier and the moisture barrier had degraded. Therefore, it is not evident that loss of material due to corrosion in inaccessible areas of the containment liner is not significant at VCSNS. Provide the following additional information:

(y) What inspections have been conducted to assess the condition of the liner embedded in the concrete base?
(z) Did the observed degradation occur before or after the implementation of the six (6) aging management programs (identified in AMR item 12) credited to preclude such degradation?
(aa) Since this type of degradation has already occurred, what is the technical basis for concluding that it could not occur again?

RAI 3.5-16

LRA Table 2.4-2 indicates that the aging management review results for numerous components of the reactor building (such as containment liner plate, cable tray, conduit, electrical and instrument panels and enclosures, fire doors, flood curbs, and HVAC duct supports), are presented in LRA Table 3.5-1, AMR item 13. LRA Table 3.5-1, AMR item 13 covers the component group "Steel elements; protected by coating," and GALL identifies the aging management program as "Protective coating monitoring and maintenance." AMR item 13 lists four aging management programs in the "Discussion" column. These are the 10 CFR 50 Appendix J General Visual Inspection, Containment Coating Monitoring and Maintenance Program, Containment ISI Program - IWE/IWL, and Maintenance Rule Structures Program. However, Report TR00170-003, Rev. 0, Attachment II, does not credit a coating monitoring and maintenance program for these components, except for the containment liner plate. Because of these inconsistencies and the grouping of so many components together within AMR item 13, it is not clear which aging management programs are being credited for which components. Therefore, clarify the following items:

(bb) Table 3.5-1, AMR item 13 covers the component group “Steel elements; protected by coating.” Are all components that reference AMR item 13 protected by coatings that are managed by a coating monitoring and maintenance program? If not, revise the LRA Table 2.4-2 “Aging Management Review Results” column.

(cc) Since the four listed aging management programs cannot apply to all components in AMR item 13, identify which VCSNS aging management programs apply to which group of components.

RAI 3.5-17

For the containment post-tensioning system, Report TR00170-003, Rev. 0, Attachment II, identifies loss of material and loss of prestress as the aging effects requiring management, and the Tendon Surveillance Program as the applicable aging management program; the match with GALL is specified as “partial”. LRA Table 3.5-1, AMR items 14 and 11 respectively address the same aging effects for the post-tensioning system, and identify the Containment ISI Program - IWE/IWL and the Tendon Surveillance Program as the applicable aging management programs. Both aging management programs are identified as consistent with GALL. To clarify this apparent contradiction, explain what is meant by a partial match in Report TR00170-003, Rev. 0, Attachment II. Also submit the technical basis for any deviations from the GALL programs that manage aging of the post-tensioning system (i.e., GALL XI.S2 and X.S1) .

RAI 3.5-18

LRA Table 3.5-1 AMR items 1, 2, and 3 discuss bellows used in containment penetrations and conclude that stress corrosion cracking (SCC) is not an applicable aging effect requiring management. The discussion under these AMR items indicates that the penetration bellows are not part of the containment pressure boundary because they are located on the exterior side of containment and hot penetrations are sealed on the inside of containment by a flat plate welded to both the penetration sleeve and process pipe. LRA Table 3.5-1, AMR item 2 states that the hot penetrations bellows “provide structural and/or functional support for process piping on the outboard side of containment; therefore, in the unlikely event of SCC in the bellows, the intended functions are not affected.” While the intended function for containment pressure boundary may not be affected, failure of the bellows would affect other intended functions. In addition, AMR items 2 and 3 credit the Appendix J General Visual Inspection, Appendix J Leak Rate Testing, and Containment ISI Program - IWE-IWL as aging management programs; these programs are only applicable to the welded flat plate closures, if the penetration bellows are not part of the containment pressure boundary. Therefore, provide the following information:

(dd) Explain why cracking of the stainless steel penetration bellows (and the associated dissimilar metal welds) does not affect the bellows’ intended function.

(ee) Identify what aging effects are applicable to the penetration bellows (and the associated dissimilar metal welds), and the aging management programs that are credited to manage aging.

RAI 3.5-19
The staff notes that some entries in the "GALL Item Number" column of Report TR00170-003, Rev. 0, Attachment II appear to be incomplete, are unclear, or have duplicate entries. Examples are

- Page 37, anchorage/embedments : for loss of material managed by the Boric Acid Corrosion Surveillances program, the "GALL Item Number" column identifies IIIB1.1.1-b and IIIB1.2.1-b. In accordance with NUREG-1801, Vol. 2, GALL Item Numbers IIIB2.1-b, IIIB3.1-b, IIIB4.1-b, and IIIB5.1-b should also be identified.
- Page 39, expansion anchors: for loss of material managed by the Boric Acid Corrosion Surveillances program, the "GALL Item Number" column identifies IIIB1.1.1-b. In accordance with NUREG-1801, Vol. 2, GALL Item Numbers IIIB1.2.1-b, IIIB2.1-b, IIIB3.1-b, IIIB4.1-b, and IIIB5.1-b should also be identified..

The applicant is requested to review and revise Report TR00170-003, Rev. 0, Attachment II, in order to ensure that the correct and appropriate GALL Item numbers are identified in the "GALL Item Number" column.

RAI 3.5-20
For the personnel airlock, escape airlock, and equipment hatch, the staff considers that loss of leak tightness in a closed position due to mechanical wear of locks, hinges, and closure mechanisms is an applicable aging effect that needs to be managed. This is NUREG-1801, Vol. 2, GALL Item Number II.A3.2-b. From the information provided on page 43 of Report TR00170-003, Rev. 0, Attachment II, it is not clear whether this aging effect will be managed for license renewal. LRA Table 3.5-1, AMR item 5 indicates an apparent commitment to manage this aging effect. However, the following statement is included in the "Discussion" column: "Operation of hatches is governed by VCSNS Technical Specifications. Plant operational experience has not identified any fretting or seal degradation. Locks, hinges, and closure mechanisms are active components; therefore, mechanical wear is not considered an aging effect." The applicant is requested to clarify its aging management review for this aging effect as follows:

(hh) Verify that loss of leak tightness in a closed position due to mechanical wear of locks, hinges, and closure mechanisms for the personnel airlock, escape airlock, and equipment hatch is an applicable aging effect requiring management.

(ii) Technical Specifications 3/4.6.1 are referenced as an aging management program in Report TR00170-003, Rev. 0, Attachment II for the personnel airlock, escape airlock, and equipment hatch. Indicate whether the application of this specification allows any deviations from the requirements specified in the GALL XI.S1, ASME Section XI, Subsection IWE. If so, describe the deviations and provide the technical basis for concluding that the aging management commitment is at least equal to the ASME Section XI, Subsection IWE aging management program.

3.5 AGING MANAGEMENT PROGRAMS

RAI 3.5-21
The Introduction to Appendix B - Aging Management Programs and Activities of the LRA states that "clarification is provided for instances where the VCSNS program does not match specific details of a NUREG-1801 program element but is still determined to be consistent." For the following aging management programs, a clarification is provided; however, it is not clear how the VCSNS program does not match the referenced GALL aging management program. Please explain what is intended by the clarification provided for each program and confirm that each program is completely consistent with GALL:

1. ASME Section XI ISI Program - IWF (B.1.13)

2. Containment ISI Program - IWE/IWL (B.1.16)

RAI 3.5-22
The staff noted several inconsistencies between the FSAR Supplement summary descriptions of the aging management programs in LRA Appendix A and the scope of the aging management programs identified in LRA Appendix B as "consistent with GALL." Some examples of these inconsistencies are:

(a) Section 18.2.5 of LRA Appendix A states that the ASME Section XI ISI Program – IWF manages "loss of material," while the parameters monitored under GALL XI.S3 are much broader and include: corrosion; deformation; misalignment; improper clearances; improper spring settings; damage to close tolerance machined or sliding surfaces; and missing, detached, or loosened support items.

(b) Section 18.2.5 of LRA Appendix A states that the ASME Section XI ISI Program – IWF manages cracking of high strength anchorage of ASME Class 1 component supports. Under GALL XI.S3 the visual inspection would be expected to identify relatively large cracks. If cracking of high strength anchorage needs to be managed, the staff would expect that the applicant would credit a program consistent with GALL XI.M18, Bolting Integrity.

For the following aging management programs identified as consistent with GALL, please verify that the complete scope of the aging management program, as described in NUREG-1801, GALL Volume 2, is being credited for license renewal aging management. If this is not the case, please identify and document the justification for each exception:

1. 10 CFR 50 Appendix J Leak Rate Testing (B.1.12)

2. ASME Section XI ISI Program - IWF (B.1.13)

3. Containment ISI Program - IWE/IWL (B.1.16)

4. Maintenance Rule Structures Program (B.1.18)

5. Service Water Pond Inspection Program (B.1.21)

6. Tendon Surveillance Program (B.3.3)

Also, revise the aging management program descriptions in LRA Appendix A to accurately reflect the scope of each program that is being credited for license renewal aging management. The descriptions should make direct reference to applicable 10 CFR sections, codes, standards, regulatory guides, and any other formal documents that define the commitment.

RAI 3.5-23
The applicant states that 10 CFR 50 Appendix J General Visual Inspection (B.1.11) is consistent with XI.S4, 10 CFR 50 Appendix J, as identified in NUREG-1801. However, the scope of GALL XI.S4 is for containment leak rate testing and not general visual inspection of containments. Inspection of containments is covered by GALL XI.S1 and XI.S2, which involve ASME Section XI, Subsections IWE and IWL, respectively. The applicant states in LRA Section B.1.16 that the Containment ISI Program - IWE/IWL is consistent with GALL XI.S1 and XI.S2. The 10 CFR 50 Appendix J General Visual Inspection (B.1.11) is included in the discussion column of LRA Table 3.5-1, but is not identified as a credited aging management program in Report TR00170-003, Rev 0, Attachment II: Aging Management Review for Structures and Structural Components.

The applicant is requested to clarify whether the 10 CFR 50 Appendix J General Visual Inspection (B.1.11) program is credited as an aging management program for license renewal and provide the following information:

(a) If it is credited, the applicant needs to verify that it supplements the Containment ISI Program - IWE/IWL for visual inspection of containment, and is not used as a substitute.

(b) If any element of the containment visual inspection relies solely on the 10 CFR 50 Appendix J General Visual Inspection (B.1.11) program, then this aging management program needs to be evaluated against the 10 program elements of an aging management program, using the guidance in Branch Technical Position RLSB-1 in Appendix A of NUREG-1800.

(c) Identify which component types listed in Report TR00170-003, Rev 0, Attachment II credit this aging management program.

RAI 3.5-24
In LRA Section B.1.12.1 on operating experience, the applicant discussed a non-conformance (NCN) that was documented for rust found on the reactor building liner plate adjacent to the moisture barrier and a degraded moisture barrier. The disposition was to clean-up the rust on the reactor building liner plate adjacent to the moisture barrier and to replace affected portions of the moisture barrier. It is stated that visual examination and ultrasonic tests demonstrated that the liner plate had not degraded. The evaluation concluded that the condition was normal surface life exposure and was not aging related.

It is unclear to the staff why the NCN discussed above was identified by the Appendix J Leak Rate Testing program (B.1.12) and not by the Appendix J General Visual Inspection program (B.1.11) and/or the Containment ISI Program - IWE/IWL (B.1.16). The staff requests the applicant to provide the following information:

(a) Confirm that this nonconformance was detected prior to the implementation of the aging management program under LRA Section B.1.16. If not, explain why this nonconformance was not detected under the B.1.16 aging management program.

(b) Explain why this nonconformance was not detected under the B.1.11 aging management program.

(c) Clarify the scope of and interaction between all three aging management programs.

(d) Explain why “normal surface life exposure” is not aging related.

(e) The rust on the liner plate and the degraded moisture barrier could indicate the presence of or result in degradation in the inaccessible areas of the containment liner. Discuss how the acceptability of the inaccessible areas of the containment liner was evaluated as a result of this nonconformance.

RAI 3.5-25
In LRA Section B.1.13.1, the applicant acknowledges that improperly heat-treated anchor bolts are susceptible to stress corrosion cracking, based on industry operating experience, but states that ASTM A490 anchor bolt material used at VCSNS is properly heat-treated by conforming to ASTM Specification A490 through a Certified Material Test Report, in accordance with station specifications. Although not stated, the applicant implies that stress corrosion cracking of A490 anchor bolt material used at VCSNS is not an aging management concern. The staff requests the applicant to specifically describe site-specific operating experience related to stress corrosion cracking of high-strength bolting materials used in Class I piping and component supports, including a description of inspection/test methods employed to detect it, and the technical basis for the adequacy of the methods employed.

RAI 3.5-26
LRA Section B.1.14 states that “although not credited for license renewal, the battery racks are also inspected for physical damage.” The staff requests the applicant to explain what is meant by “physical damage” and how this is distinguished from structural damage or degradation.

RAI 3.5-27
The Flood Barrier Inspection Program described in LRA Section B.1.17 is included in the discussion column of LRA Table 3.5-2, but is not credited for license renewal in Report TR00170-003, Rev 0, Attachment II: Aging Management Review Results for Structures and Structural Components. The staff requests that the applicant provide the following information regarding this program:

(a) Clearly state the component types and associated structures that credit this program for license renewal.

(b) Explain the added value of this program since LRA Section B.1.17 states that either the Fire Protection Program or the Maintenance Rule Structures Program manages all flood barrier components.

(c) Clarify why the scope section of this program indicates that there are flood seals in the intermediate building. The staff notes that flood barriers are not identified as a component type for the intermediate building in Report TR00170-003, Rev 0, Attachment II.

(d) The section on “monitoring and trending” states the frequency of inspection for flood barrier seals that are also fire barrier penetration seals. Provide the frequency of inspection for flood barrier seals that are not fire barrier penetration seals, as well as all the other components within the scope of this program, such as flood barriers (walls, curbs, equipment pedestals) and flood doors.

RAI 3.5-28
LRA Section B.1.18 states that the Maintenance Rule Structures Program is consistent with GALL XI.S6 with several listed enhancements that will be incorporated into the program prior to the period of extended operation. The staff requests that the applicant provide the following information regarding this program:

(a) Verify that the scope of this program includes visual inspection of concrete for aging effects of loss of material, cracking and change in material properties and explain what this program requires for VCSNS concrete structures.

(b) Since the North Berm, an earthen embankment, will be incorporated into the scope of this program, clarify that this program is also completely consistent with all the attributes of GALL XI.S7, RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants.

(c) Since this program is credited for managing aging effects of masonry walls, clarify that this program is also completely consistent with all the attributes of GALL XI.S5, Masonry Wall Program.

(d) Clarify the apparent editorial mistake in the last sentence of the second paragraph of LRA Section B.1.18.1 that states: “… including the protection and support of 0 systems and components.”

(e) Clarify whether the commitment to incorporate the enhancements to this program discussed in LRA Section B.1.18 will also be included in the FSAR Supplement, Appendix A, Section 18.2.22. This section does not currently include such a commitment.

RAI 3.5-29
The Pressure Door Inspection Program described in LRA Section B.1.20 is included in the discussion column of LRA Table 3.5-2, but is not credited for license renewal in Report TR00170-003, Rev 0, Attachment II: Aging Management Review Results for Structures and Structural Components. The staff requests that the applicant provide the following information regarding this program:

(a) Clearly state the component types and associated structures that credit this program for license renewal.

(b) Under “parameters monitored or inspected” it is stated that “Excessive wear for door appurtenances such as latches, gaskets, hinges, sills, and closing devices are additional attributes in the technical requirements package, but are not credited for license renewal.” However, LRA Appendix A, Section 18.2.24 states that “Pressure door inspection attributes include freedom of movement, function (closed during normal plant operation), structural deterioration, and loss of door/door hardware material.” These inconsistencies should be clarified.

(c) Under “monitoring and trending,” provide the frequency of inspection for all pressure doors within the scope of this program.

RAI 3.5-30
LRA Section B.1.21 states that the Service Water Pond Dam Inspection Program is consistent with GALL XI.S7 with several listed enhancements that will be incorporated into the program prior to the period of extended operation. The staff requests that the applicant provide the following information regarding this program:

(a) Clarify whether the commitment to incorporate the enhancements to this program discussed in LRA Section B.1.21 will also be included in the FSAR Supplement, Appendix A, Section 18.2.31. This section does not currently include such a commitment.

(b) The discussion in LRA Section B.1.21.1 on operating experience does not include the East Dam. Please provide a discussion on the operating experience for the East Dam.

RAI 3.5-31
LRA Section B.1.23 Underwater Inspection Program (SWIS and SWPH), states that the scope of the program includes underwater inspections of both the service water intake structure (SWIS) and the service water pump house (SWPH). Report TR00170-003, Rev 0, Attachment II, states that the Underwater Inspection Program is credited for managing the aging effects for both SWIS and SWPH components for (1) loss of material and cracking in a raw water environment for concrete materials, and (2) loss of material in a raw water environment for steel materials. The concrete components identified in Attachment II include intake bays or canals and reinforced concrete - beams, columns, floor slabs, walls. The steel components identified in Attachment II include intake screens. The staff notes that the discussion column of LRA Table 3.5-2 states that VCSNS uses the Service Water Pond Dam Inspection Program (which is stated to be consistent with GALL XI.S7) inspections only for supplementary review for both the SWIS and SWPH. In order to complete the evaluation of this program, the staff requests that the applicant provides the following information:

(a) It is the staff’s position that an effective aging management program for water control structures should incorporate the attributes described in GALL XI.S7. Since the applicant uses the Service Water Pond Inspection Program for supplementary review, the staff requests that the applicant explain which attributes from this program are not used for the inspections performed under the Underwater Inspection Program and provide a technical bases for their omission.

(b) Several aging management program attributes discussed in LRA Section B.1.23 focus mainly on the SWIS. The applicant is requested to discuss the following AMP attributes as they apply to the SWPH components identified in Report TR00170-003, Rev 0, Attachment II:

i. parameters monitored or inspected
ii. monitoring and trending
iii. acceptance criteria
iv. operating experience

(c) With regard to the section on “Detection of Aging Effects,” explain what is meant by the expression “attributes associated with aging” for both the SWIS and SWPH.

(d) It is the staff’s understanding that the complete scope of the Underwater Inspection Program is performed every five years for the SWIS. Please confirm that the staff’s understanding is correct and that the inspection frequency also applies to the SWPH.

(e) The description of the Underwater Inspection Program for the FSAR Supplement in LRA Appendix A, Section 18.2.38 implies that underwater inspections of the SWPH only serve to monitor corrosion and fouling within the Service Water System. If this is not correct, describe how the FSAR Supplement will be modified to reflect the complete scope of this program as it applies to the SWPH. If the scope of the program is limited as the statement implies, explain how the program can be credited for managing the SWPH aging effects discussed in the first paragraph of this request.

(f) The conclusion provided in LRA Section B.1.23.2 states that “the Underwater Inspection Program (SWIS and SWPH) has been demonstrated to be capable of detecting and managing the effects of aging for concrete components in fluid environments.” Please clarify why this conclusion omits reference to the aging effects for steel materials such as the intake screens.

RAI 3.5-32

In LRA Section B.3.3, the applicant states that a review of the non-conformances (NCNs) written to address programmatic and problematic deficiencies with the Tendon Surveillance Program indicates that there have been no adverse trends associated with aging that are not inherent to this type of post tensioning system.

The applicant states that a non-conformance (NCN) was identified to address the collection of water due to in-leakage into the auxiliary building tendon sump area to a depth that submerged a tendon end cap. The water level in the pit was reduced to a level below the tendon end cap. During RF-12 the tendon end cap was removed for inspection and no free water was found. Grease samples (analyzed for entrained moisture) and the tendon components (inspected for corrosion) were found to be acceptable. As a corrective action, Operations added the auxiliary building tendon sump area to their trend logs and will request facilities to drain the area if the water level in the area approaches the level of the tendon end cover.

The staff has concerns about the long-term condition of the tendon anchorages if subjected to additional episodes of water infiltration. Such environments could potentially degrade the tendon anchorage system, including anchor components inside the end cap, the baseplate and reinforced concrete region around the anchors. The staff requests the applicant to (1) explain the relationship between the auxiliary building tendon sump area and the tendon access gallery beneath the containment; (2) identify the type of tendon end caps (horizontal, vertical) in the auxiliary building tendon sump area; (3) identify whether the tendon access gallery is also included in the Operations “trend logs” to prevent excessive water level, and if not, explain why not; (4) discuss whether this commitment is credited for management of aging of the tendon prestressing system; and (5)discuss why water is allowed to remain in the auxiliary building tendon sump area and only drained if the water level in the area approaches the level of the tendon end cover.

4.5 CONCRETE CONTAINMENT (REACTOR BUILDING) TENDON PRESTRESS TLAA

RAI 4.5-1
Section 4.5 of the LRA indicates that the reactor building tendons are a TLAA, and VCSNS will utilize 10 CFR 54.21(c)(1) - Option (iii) to demonstrate that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation. Appendix B.3.3 of the LRA indicates that the Tendon Surveillance Program is consistent with X.S1, Concrete Containment Tendon Prestress, as identified in NUREG-1801. In order for the staff to determine the adequacy of the tendon prestressing force and the TLAA for the period of extended operation, an understanding of the past operating experience for the tendons is needed.

Test results from the first three surveillances indicated that the wire relaxation force losses in the tendon system were greater than that which were predicted during design (resulting in lower measured prestressing forces). Therefore, in June 1988, the predicted wire relaxation force losses were increased from 8.5% to 12.5%. Then in the fourth period (10^{th} year) tendon surveillance, the vertical tendons were retensioned because the previous surveillance data indicated that the vertical tendon forces would be below the Technical Specifications minimum prior to the fifth period surveillance. Although the fifth period (15^{th} year) and sixth period (20^{th} year) tendon surveillances have been completed, no information was provided regarding the comparison of the measured tendon forces to the predicted lower limit at the 15^{th} and 20^{th} year tendon surveillances. LRA Section 4.5 indicates that based on trending data and results from previous surveillances, “VCSNS does not currently expect the tendons to provide adequate prestress for 60 years without future retensioning of various members.”

In order to make a reasonable assessment regarding the effectiveness of the TLAA, the staff requests that the applicant provide the following information:

(e) Based on the past operating experience, provide the comparison of the measured lift-off forces against the predicted lower limits and minimum required values for each group of tendons, and project the data through the period of extended operation. These curves should reflect the past retensioning of the tendons. Provide the trend lines based on the VCSNS Tendon Surveillance Program for each group of tendons, showing measured prestressing forces above and below the trend lines. Identify whether the guidance in Information Notice 99-10 is implemented.

(f) Provide a brief description of the reason why the tendon wire relaxation values were greater than those used in the design of the tendon system.

Virgil C. Summer LRA
Request for Additional Information

2.4.1-D1 Section 2.4.1, “Reactor Building”, of the LRA states that the Reactor Building consists of a cylindrical wall, a shallow dome roof and a foundation mat with a depressed incore instrumentation pit under the reactor vessel and the foundation mat bears on fill concrete that extends to competent rock. Table 2.4-2, ”Reactor Building Component Types Subject To Aging Management review and Their Intended Functions”, lists “Foundations” as a component type. Since Table 3.5-1,Item 9, reduction in foundation strength due to erosion of porous concrete subfoundation, was listed as an AMR result for the foundations in Table 2.4-2, the staff interprets that your designation “foundations” include the foundation mat and the fill concrete, which is the subfoundation. Verify whether the staff’s interpretation is correct. If not, state what the foundations consist of. The Auxiliary Building, Control Building, Fuel Handling Building, Intermediate Building, Turbine Building, and Service Water Discharge Structure are also supported on fill concrete, and “Foundations” is also listed as a component type for them. However, Table 3.5-1,Item 9, is not listed as an AMR result for these buildings. Clarify why Table 3.5-1,Item 9, is listed as an AMR result only for the Reactor Building but not for other buildings whose foundations are also supported on a fill concrete subfoundation.

2.4.1-D2 Section 2.4.1.3, “Penetrations”, states that a fuel transfer tube penetrates the Reactor Building connecting the refueling canal in the Reactor Building and the fuel transfer canal in the Fuel Handling Building. The staff finds that the fuel transfer tube is not listed in Table 2.4-2 as a component requiring AMR. If the fuel transfer tube requires an AMR, indicate where it is listed. If not, present a justification for not requiring an AMR.

2.4.1-D3 Section 2.4.1.3, “Penetrations”, states that double O-rings are used to seal the doors of two personnel airlocks and an equipment hatch and they are not considered as long-lived components because they are tested and replaced when warranted by their condition, and therefore do not require an AMR. 10 CFR 54.21 (a)(1)(ii) states that a component, which is not subject to replacement based on a qualified life or specified time period, is subject to an AMR. Since the O-rings may fail in the period of time between tests and you did not indicate that the O-rings have a specified time period for replacement, provide a justification that the O-rings meet the requirement of 10 CFR 54.21 (a)(1)(ii).

2.4.1-D4 Table 2.4-2, ”Reactor Building Component Types Subject To Aging Management Review And Their Intended Functions”, lists “Anchorage”, “Anchorage / Embedments (exposed surfaces)”, and “Embedments” as component types requiring AMR. Define or describe each component type so that a clear understanding on what each component type represents and on what are the distinction or difference among the three component types.

2.4.1-D5 Indicate whether there are masonry block walls in the Reactor Building which are subject to AMR.

2.4.2-D1 Section 2.4.2.1, “Auxiliary Building”, of the LRA states that “the Hot Machine Shop is a steel framed building with metal siding designed to withstand earthquake loads and tornado wind loads to the extent required for prevention of damage to Seismic Category I structures.

The north wall of the Auxiliary Building is separated from the Hot Machine Shop by a seismic gap. The failure of the Hot Machine shop will not prevent the satisfactory accomplishment of any required safety-related functions. The Hot Machine shop is therefore not subject to an aging management review." Does your statement indicate that the Hot Machine Shop was so designed that it will not collapse under earthquake loads and tornado wind loads or that it may collapse but it will not impact on, or be in contact with, Seismic Category I structures? Does the word "failure" in your statement include the collapse of the Hot Machine shop? If not, define the kind of failure. Your statement appears to be a reason for excluding the Hot Machine Shop to be in scope, but not for excluding it from an AMR as you stated. Table 2.2-2: Structural Scoping Results of the LRA lists the Hot Machine Shop as in scope, and the reason for being in scope is listed as that its intended functions are those that meet the requirements of 10 CFR 54.4(a)(2), which is a seismic II/I concern. Your statement in Section 2.4.2.1 appears to be contradictory to the intended functions listed in Table 2.2-2. Clarify whether the Hot Machine Shop should be in scope and requiring an AMR, and provide a justification for your determination.

2.4.2-D2 Section 2.4.2.1, "Auxiliary Building", of the LRA states that the southwestern portion of the Auxiliary Building supports two large tanks, the Refueling Water Storage Tank and the Reactor Make-up Water Storage tank. The staff finds that these two tanks are not listed in Table 2.4-3: Auxiliary Building Component Types Subject To Aging Management Review And Their Intended Functions. Indicate whether these two tanks are subject to an AMR. If yes, indicate the table number or location where they are listed. If not, provide a justification for not requiring an AMR.

2.4.2-D3 The staff finds that you did not list grout as a component that requires an AMR in Section 2.4. Indicate whether grout is subject to an AMR. If yes, indicate the table number or location where it is listed. If not, provide a justification for not requiring an AMR.

D-RAI 3.4-1
In Section 2.3.4.10 of the LRA, the applicant lists Turbine Cycle Sampling System components subject to aging management review but Section 3.4.1, which lists VCS Steam and Power Conversion System does not include the Turbine Cycle Sampling System. Should the list of systems in Section 3.4.1 include the Turbine Cycle Sampling System?

D-RAI 3.4-2 For the Steam and Power Conversion System (SPCS) in LRA Tables 2.3-38 thru 2.3-47, the applicant does not identify any SPCS components that are managed for cumulative fatigue. NUREG-1801 recommends aging management of cumulative fatigue for piping and fittings in the main steam, feedwater, and auxiliary feedwater systems. Explain why Tables 2.3-38 thru 2.3-47 do not identify any SPCS components that are managed for cumulative fatigue.

D-RAI 3.4-3
LRA Table 3.4-1, item 1, identifies the applicant's aging management for cumulative fatigue damage for piping and fitting in the main feedwater line, the steam line, and for AFW piping. In the discussion column for this item, the applicant states, "see Section 4.3.2 [of the LRA] for the TLAA discussion of Class 2 and 3 piping." The discussion column does not state if the applicant's TLAA is consistent with the NUREG-1801 TLAA program. For the SPCS piping, NUREG 1801 recommends an evaluation of allowable stress levels based on the number of anticipated thermal cycles as described in NUREG-1800, Section 4.3.1.1.2. Does the applicant perform the thermal cycle evaluation of SPCS piping as described in NUREG-1800, Section 4.3.1.1.2 for the main feedwater line, the steam line, and for AFW piping? If so, is the applicant's TLAA program consistent with NUREG-1801. If not, please explain any differences.

D-RAI 3.4-4
In Table 3.4-1, item 2 of the LRA, the applicant states that various components will be managed for the aging effect of loss of material due to general (carbon steel only), pitting, and crevice corrosion using the applicant's Chemistry Program but a One-Time Inspection is not warranted to verify corrosion is not occurring for components in this group, except for the condensate storage tank. The applicant further states that a review of operating experience confirms the effectiveness of the Chemistry Program for treated water to manage aging effects when continued into the period of extended operation. NRC staff position is that a one-time inspection is needed to address concerns for the potential long incubation period for certain aging effects on structures and components. There are cases where either (a) an aging effect is not expected to occur but there is insufficient data to completely rule it out, or (b) an aging effect is expected to progress very slowly. For these cases, there needs to be confirmation that either the aging effect is indeed not occurring, or the aging effect is occurring very slowly as not to affect the component or structure intended function. A one-time inspection of select components and susceptible locations is an acceptable method to ensure that corrosion is not occurring and that the component's intended function will be maintained during the period of extended operation. The one-time inspection should be performed late in the current operating period to ensure aging effects will not affect the component intended function during the period of extended operation. Also, this RAI applies to the valves in Table 3.4-2, item 5. The applicant is requested to perform a one-time inspection of components in this group using

the criteria stated in NUREG-1801, XI.M32, “One-Time Inspection.” The one-time inspection should be based on severity of conditions, time of service, and lowest design margin. Otherwise, the applicant should explain how operating experience will ensure the integrity of components during the period of extended operation.

Note: NUREG-1801 does not recommend a One-Time Inspection for main steam system piping. Therefore, a one-time inspection is not needed to verify the Water Chemistry Program for main steam system components in LRA Table 2.3-44.

D-RAI 3.4-5
In Table 3.4-1, item 2 of the LRA, the applicant does not include components such as feedwater pumps, condensate storage tank piping (only the tanks), condensate cleanup system, or blowdown system pumps & Hx. Are any of these components in-scope & require AMR for license renewal. If yes, explain why the chemistry program is not used to mange loss of material due to general (carbon steel only), pitting, and crevice corrosion.

D-RAI 3.4-6
Loss of material due to general corrosion, pitting and crevice corrosion, microbiologically influenced corrosion (MIC), and biofouling could occur in carbon steel piping and fittings for untreated water from the backup water supply in the auxiliary feedwater system. The applicant must provide reasonable assurance that these aging effects are adequately managed. In Table 3.4-1, item 3 of the LRA, the applicant does not manage raw water exposure to AFW piping. In the discussion column, the applicant states that the “AFW piping at VCSNS is not exposed to untreated water. The Service Water System provides emergency backup to the Emergency Feedwater System through automatic isolation valves that normally provide boundary isolation between the treated water of the Emergency Feedwater System and the untreated water of the Service Water System.” Please explain what is meant by the statement that, “automatic isolation valves that normally provide boundary isolation," and how the applicant has verified that the AFW piping has not been exposed to raw water.

D-RAI 3.4-7
In Table 3.4-1, item 4 of the LRA, the applicant’s aging management review for Auxiliary Feedwater system pump lubricating oil coolers determined that water and contaminates will not intrude into the oil environments for these components. The staff’s position is that an environment of lubricating oil contaminated with water may cause loss of material of carbon or stainless steel heat exchanger components due to general corrosion (carbon steel only), pitting, crevice corrosion and microbiological influenced corrosion. On this basis, the Auxiliary Feedwater system pump lubricating oil coolers have the potential of being contaminated with water. Also, this RAI applies to the heat exchangers in Table 3.4-2, item 3. Explain why water and contaminates will not intrude into the oil environments for these heat exchangers and why oil samples are not credited to ensure water does not contaminate the lube oil.

D-RAI 3.4-8
In Table 2.3-46 of the LRA, the applicant identifies that the Turbine Cycle Sampling System pipe and valves are managed for aging by the AMP B.2.1, “Inspection for Mechanical Components.” The scoping section of AMP B.2.1 identifies the mechanical systems managed by the AMP but does not include the Turbine Cycle Sampling System. Explain why the Turbine Cycle Sampling System is not included in the scope section of AMP B.2.1.

D-RAI 3.4-9
LRA Tables 2.3.38 thru 2.3.47, the applicants lists “valve body” in the component column. NRC position is that the aging effects identified in these tables, except for wall thinning due to flow-accelerated corrosion, are applicable to both the valve body and bonnet. Please explain why the valve bonnets are not affected by these aging effects or provide aging management for the bonnets.

D-RAI 3.4-10
In Table 3.4-1, item 6 of the LRA, the applicant states in the discussion column for its Flow Accelerated Corrosion Program that “ the component/component type AMR results for VCSNS are consistent with NUREG-1801 in material, environment, aging effects, and program. In NUREG-1801, aging management for Flow Accelerated Corrosion is specified for all Steam and Power Conversion System piping, fitting, pump casings, and valve bodies. In Tables 2.3-38 and 2.3-40, the applicant does not specify aging management for Flow Accelerated Corrosion for piping, fitting, pump casings, and valve bodies in the Auxiliary Boiler and Feedwater System and the Emergency Feedwater System. Also, in Table 2.3-44, the applicant does not specify aging management for Flow Accelerated Corrosion for the Main Steam System pump turbine (casing only). Please explain why the applicant states it is consistent with NUREG-1801 but does not include the above components in its FAC program.

D-RAI 3.4-11
The main objective of the Water Chemistry Program is to mitigate damage caused by corrosion and stress corrosion cracking. NUREG-1801 recommends implementation of the Water Chemistry Program to manage loss of material due to general (carbon steel only), pitting, crevice, and stress corrosion cracking (stainless steel only) for piping and fittings, valve bodies and bonnets, pump casings, tanks, tubes, tubesheets, channel head, and shell. The applicant does not credit its Chemistry Program for the following components: 1) condensate storage tank in Table 2.3-39, 2) Heat exchanger tube and shell in Table 2.3-40, 3) tank (reservoir) in Table 2.3-40, and 4) pump turbine (case only) in Table 2.3-44. Explain why the Chemistry Program is not credited to manage loss of material due to general (carbon steel only), pitting, crevice, and stress corrosion cracking (stainless steel only) for these components.

D-RAI 3.4-12
NUREG-1801 recommends that heat exchanger internal exposed to raw or treated water be managed for loss of material by the Open Cycle and Closed Cycle Cooling Water System AMPs. In Table 3.4-1, items 9 & 10 of the LRA, the applicant states that the Open Cycle and Closed Cycle Cooling Water System AMPs as described in NUREG-1801 are not used in any Steam and Power Conversion Systems at VCSNS. Are there any Steam and Power Conversion System heat exchangers at VCSNS exposed to raw or treated water that require aging management review? If yes, identify the heat exchangers, the aging effects, and how the aging effects are managed?

D-RAI 3.4-13
In Table 3.4-1, item 12 of the LRA, the applicant states that the Inspection of Mechanical Components is used to monitor the external surfaces of the above ground Condensate Storage Tank for loss of material. For tanks supported on earthen or concrete foundations, corrosion my occur at inaccessible locations, such as the tank bottom. Is the bottom of the

Condensate Storage Tank located on an earthen or concrete foundation? If so, are thickness measurements taken at the tank bottom to ensure that significant degradation is not occurring and the component intended function will be maintained during the extended period of operation.

D-RAI 3.4-14

In Table 3.4-1, item 12 of the LRA, the applicant states that there is underground piping in the AFW (Emergency Feedwater System at VCSNS) and that the Buried Pipe and Tanks Inspection program will manage the aging effects. LRA Table 2.3-40, for the Emergency Feedwater System, only identifies orifices as subject to aging management by the Buried Pipe and Tanks Inspection program. Explain why the AFW piping in Table 2.3-40 does not refer to the Buried Pipe and Tanks Inspection program and how the underground piping in the AFW is managed for aging.

V.C. Summer LRA Engineered Safety Features Systems
Request for Additional Information

RAI 3.2-D1
In LRA Section 2.3.2, "Engineered Safety Features", Reactor Building Spray System is listed as one of the seven subsystems that make up the ESF system. However, in Table 3.2-1, Item 2, the applicant mentioned VCSNS Containment Spray System, instead. The applicant is requested to clarify this mistake.

RAI 3.2-D2
In LRA Table 3.2-1, Item 3, the applicant stated that VCSNS has determined that loss of material of the underside of the refueling water storage tank (RWST) is not an aging effect requiring management as this stainless steel tank is not buried. The staff is not clear as to how the applicant arrived at such conclusion. The applicant is requested to discuss the potential corrosive environments that may surround the tank bottom, and justify the determination of no aging effects requiring management.

RAI 3.2-D3
In LRA, Table 3.2-1, Item 4, the applicant stated that MIC has been determined not to be a valid aging effects/mechanisms for the material/environment combination represented by the containment isolation valves and associated piping. The applicant stated that this is because the four systems (AC, DN, LR, NG) that provide containment isolation are not subject to wetting from raw water. The applicant is requested to delineate the details of the physical environments that are associated with the containment isolation valves and associated piping, for each of the four systems, and justify that the components are not susceptible to the aging effect of MIC.

RAI 3.2-D4
In LRA Table 3.2-2, Item 1, the applicant stated that sheltered environments do not contain contaminants of sufficient concentration to cause aging effects require aging management, for the stainless steel components. The applicant is requested to provide the basis, for all the variety of potential sheltered environments in ESF, that stainless steel components are not susceptible to any aging effects requiring management.

RAI 3.2-D5
In LRA Table 3.2-1, Item 5, the applicant stated that, for high-pressure safety injection pump mini-flow orifice, the aging effect of loss of material due to erosion is considered a design problem, and, therefore, does not have an identified aging management program for erosion of mini-flow orifices. The applicant is requested to provide a discussion on such design problem, and the procedure in place to resolve the design problem.

RAI 3.2-D6
In LRA Table 3.2-1, Item 9, the applicant stated that, for components serviced by closed-cycle cooling system, the Chemistry Program (Appendix B.1.4) has proven effective in maintaining systems' chemistry and detecting abnormal conditions. The applicant also stated that a review of operating experience confirms the effectiveness of the Chemistry Program to manage aging effects when continued into the period of extended operation. The applicant, therefore, concluded that a verification program, such as one-inspection, is not warranted for the

components in this component group. The staff notices that although operating experience is a valuable indicator for a plant's fitness to continue its operation in the extended period, industry experience may not support using it as a substitute for a verification program, without adequate justification. This is especially true for the susceptible locations of a system which, because of slow and stagnant flow conditions, requires the verification of the absence of loss of material and the effectiveness of the Chemistry Program. The applicant is requested to ensure that the intent of the GALL's one-time inspection is met.

RAI 3.2-D7
In LRA Table B-1, the applicant noted that Bolting Integrity Program of GALL is not credited for aging management. In LRA Tables 2.3-8 through 2.3-17, the applicant did not list closure bolting as a separate component type requiring AMR review, for engineered safety features systems. Also, in Table 3.2-1, Item 12, the applicant states that loss of mechanical closure integrity is not considered an aging effect requiring evaluation for Non-Class 1 component bolted closures within the scope of license renewal at VCSNS. The applicant further stated that the bolting/fasteners within the scope of license renewal were not itemized as a separate Non-Class 1 component/component type. Rather, bolting was treated as a "piece-part" (or sub-component/sub-part) of Non-Class 1 components/component types. The staff finds the above statements made by the applicant unacceptable, and would request the applicant to do the following:

1. Provide the basis of not considering loss of mechanical closure integrity an aging effect,

2. Discuss what aging effects/mechanisms have been identified for the closure bolting, even if it is treated as a sub-component, and Considering that closure bolting and its associated component may be different in material, environmental exposure, and potential mode of failure,

3. The applicant is requested to discuss how the plant-specific aging management program of closure bolting, when treated as a "piece-part" (or sub-component/sub-part) of Non-Class 1 components/component types, is measured against the intent of XI.M18, "Bolting Integrity", of GALL. The applicant is requested to ensure that all the attributes of the plant-specific program meet the intent of the corresponding GALL Chapter XI program attributes.

B.1.2 Boric Acid Corrosion Surveillances (BACS) Program

RAI B.1.2-1
The definition section of VCSNS LRA Appendix B.1.2 for "Boric Acid Corrosion Surveillances" (BACS) states that the AMP is consistent with GALL XI.M10, except for enhancements related to dissimilar metal weld inspections. The staff requests the applicant to provide more information on the following:

- operating experience (1) since publication of GALL report, and (2) since submittal of the application and the impacts of this experience on the program, if any.
- how the systems outside containment, currently inspected under other existing programs, will continue to be inspected under the enhanced Boric Acid Corrosion Surveillances Program

RAI B.1.2-2
The VCSNS LRA Appendix B, Section B1.2.1 "Boric Acid Corrosion Surveillances" - Operating Experience mentions the hot leg axial cracking at VCSNS on 10/07/00 and then states that the Boric Acid Corrosion Surveillances were subsequently enhanced to ensure that all dissimilar metal welds are included in the population of components that are visually inspected at refueling outages or when appropriate plant conditions permit access. What is the location of other dissimilar metal welds exposed to borated coolant?

RAI B.1.2-3
LRA Table 3.1-1, AMR Item 26, credits the boric acid corrosion surveillances program (LRA Appendix B.1.2) for managing loss of material due to boric acid corrosion of pressurizer carbon steel and low alloy steel components: shell, upper and lower heads, nozzles, integral support, and manway cover and bolts. Provide detailed information showing how the program will be sufficient to manage the corrosive effects of boric acid leakage on these components during the extended period of operation, including postulated leakage from the pressurizer nozzle-to-vessel welds, pressurizer nozzle-to-safe end welds, and pressurizer manway bolting materials.

B.1.4 Chemistry Program

RAI B.1.4-1
As stated in the program description, the Chemistry Program "does not commit to performing one-time inspections to verify the effectiveness of the Chemistry Program as suggested by NUREG-1801 under XI.M2." The following list provides examples of components exposed to chemically treated water that will be managed by the Chemistry AMP alone:

- Table 3.1-2, AMR Item 3, loss of material due to crevice, general, pitting, and galvanic corrosion in CS SG components (other than the shell - upper and lower barrel, transition cone, elliptical head)

- Table 3.1-2, AMR Item 5, crevice and pitting corrosion of SS piping and piping system components

- Table 3.1-2, AMR Item 6, crevice and pitting corrosion of piping and piping system components

- Table 3.1-2, AMR Item 7, crevice and pitting corrosion of reactor internals, reactor vessel, RCP, incore thermocouple seal

- Table 3.3-1, AMR Item 14, loss of material in RCP thermal barrier flange

- loss of material in SS and Ni-alloy reactor vessel components (i.e., CRD housings, cladding, vent plug, bottom head and closure head penetration tubes, reactor vessel core support pads, and nozzle safe ends)

- loss of material and cracking on the outside surface of the BMI thimble tubes and the inside surface of the guide tubes supporting the thimble tubes between the seal table and vessel lower head

- loss of material in carbon steel components in the air handling and local ventilation and cooling, chilled water, spent fuel pool cooling, and gaseous waste processing systems due to crevice, galvanic, general, and pitting corrosion

The staff requests the applicant to provide a more detailed discussion of operating history and the results of the most recent surveillances / inspections for these and similar components in the various water environments to provide assurance that the above-listed types of corrosion (crevice, general, pitting, and galvanic) do not occur. Is there a one-time inspection for the most susceptible locations (such as low flow or stagnant areas) in the systems that credit this program?

RAI B.1.4-2
Since the applicant has combined aspects of several GALL programs into its chemistry program, the staff requests the applicant to discuss to what extent this program relies on the GALL AMPs described in Chapters XI.M20, “Open-Cycle Cooling Water System,” and XI.M21, “Closed-Cycle Cooling Water System. In addition, the applicant is requested to discuss how the features of these GALL programs are incorporated into the VCSNS chemistry and cooling water corrosion programs. References to the various EPRI documents for the chemistry guidelines credited in this program should also be stated in the FSAR supplement.

RAI B.1.4-3
The table entitled “Virgil C. Summer Nuclear Station Database AMR Query,” provided by the applicant in a letter to the NRC dated October 1, 2002, states in different locations that CS components (such as CS cooling coil headers or pump casings, evaporator tubesheets and water boxes, valve bodies, pipe and fittings, and tanks) in a treated water environment are subject to cracking due to stress corrosion cracking (SCC). More information is provided in the accompanying applicant-supplied Database AMR Query Notes A-CC-a and A-CC-b. However, no aging management program has been credited to manage this aging effect. SCC occurs in carbon steels usually in the presence of hydroxides, carbonates or nitrates according to the ASM Handbook, Vol. 11, "Failure Analysis and Prevention,” and EPRI TR-107396, October 1997, Closed Cooling Water Chemistry Guideline. The staff requests the applicant to

explain how the Chemistry Program (or another specified program) can prevent, detect or mitigate the effects of SCC in these carbon steel components.

RAI B.1.4-4
The applicant states that the Chemistry Program is consistent with GALL XI.M30, “Fuel Oil Chemistry,” however, the GALL discusses verification of the program’s effectiveness at locations where contaminants may accumulate. Therefore, the staff requests the applicant to provide a basis for not including the verification of this program to manage loss of material.

RAI B.1.4-5
RAI 3.1.2.4.6-7 In LRA Table 3.1-2, AMR Item 7, the applicant credits the chemistry program (LRA Appendix B.1.4) for managing loss of material due to crevice and pitting corrosion in pressurizer shell and heads clad with austenitic stainless steel and stainless steel components internally exposed to chemically treated borated coolant. These components are susceptible to crevice and pitting corrosion because high levels of oxygen may be present in PWR reactor coolant. However, PWR licensees maintain hydrogen overpressure in the reactor coolant, and if the overpressure were at sufficiently high levels, it would provide protection in creviced geometries on the internal surfaces of the pressurizer. Explain how the chemistry program will provide for a sufficient level of hydrogen overpressure to manage crevice corrosion of the internal surfaces of the pressurizer.

B1.15 Containment Coating Monitoring and Maintenance Program

RAI B1.15-1
The Containment Coating Monitoring and Maintenance Program, as described in LRA FSAR Section 18.2.11, states that, for inaccessible areas, sampling approaches based on plant-specific characteristics, industry-wide experience, and testing history are evaluated in lieu of actual visual inspections. The staff requests the applicant to provide additional details of these sampling procedures to verify that aging-related degradation of the containment coating will be effectively managed in accordance with the current licensing basis during the period of extended operation. In addition, the staff requests the applicant to provide additional information on the confirmatory element 4 (Detection of Aging Effects), of the Containment Coating Monitoring and Maintenance Program consistent with the SRP-LR and in sufficient detail to allow adequate assessment of this element. If for some reason, this element was determined to not be applicable, provide a justification for this determination.

B.1.6 Flow-Accelerated Corrosion Monitoring Program

RAI B.1.6-1
The applicant stated that the Flow-Accelerated Corrosion (FAC) Monitoring Program is consistent with the ten elements of the Aging Management Program XI.M17, “Flow-Accelerated Corrosion,” specified in the GALL report. The applicant notes in the LRA that the need for inspections is determined by a calculation performed in accordance with engineering procedures and states that if components exhibit high wear during a cycle they are replaced with more FAC-resistant material. The EPRI document, NSAC-202L-R2, “Recommendations for an Effective Flow-Accelerated Corrosion Program,” recommends use of a predictive method for determining the rate at which component degradation by FAC is occurring. The NRC staff notes that CHECWORKS or a similar predictive code should be used to predict

component degradation in the systems conducive to FAC, as indicated by specific plant data, including material, hydrodynamic, and operating conditions. The applicant is asked to provide more detail about the “calculation performed in accordance with engineering procedures” to determine inspection need. What methods are used at VCSNS for predicting component degradation by FAC? How are these predictive methods used to determine the need and frequency of inspections?

RAI B.1.6-2
In order for the staff to evaluate the acceptability of the Flow Accelerated Corrosion (FAC) program, the applicant should provide a list of the components in the program most susceptible to FAC. The list should include initial wall thickness (nominal), current (measured) wall thickness and the future predicted wall thickness.

RAI B.1.6-3
The Flow Accelerated Corrosion (FAC) program in the applicant’s plant includes prediction of the wall thinning for the components susceptible FAC. In order to allow the staff to evaluate the accuracy of these predictions, the applicant should provide a few examples of the components for which wall thinning is predicted by the code and at the same time measured by UT or any other method employed in the applicant’s plant. This procedure will show the effectiveness of the applicant’s predictive code in predicting the as-found condition.

B.2.1 Above Ground Tank Inspection

RAI B.2.1-1
The Above Ground Tank Inspection description states in LRA Section B.2.1 under element 5 (“Monitoring and Trending”) that no actions are taken to trend inspection results. This is a one-time program used to determine if further actions are required. The NRC staff notes that the evaluation of appropriateness of the techniques and timing of the one-time inspection improve with the accumulation of plant-specific and industry-wide experience. Will the applicant be compiling and evaluating such information? As a result of the insights gained from the recent discovery of boric acid-induced corrosion of the Davis-Besse vessel, the staff requests that the applicant address the changes that may be made in monitoring and trending (considering that certain components are exposed to borated water) in response to the Davis-Besse event.

RAI B.2.1-2
The staff notes that GALL Program XI.M29 “Above Ground Carbon Steel Tanks,” which defines preventive measures to mitigate corrosion by protecting the external surface of carbon steel tanks with paint or coatings in accordance with standard industry practice, is not credited for aging management in the VCSNS LRA. The applicant is asked to provide evidence that LRA Section B.2.1 adequately addresses above ground tank inspection in its stipulation of addressing only the internal surfaces of tanks. The staff notes that Section B.1.15 of the LRA describes the Containment Coating Monitoring and Maintenance Program as an existing aging management program that manages loss of material due to coating degradation and is not credited with managing degradation of tanks (LRA Section 2.3). Is the external corrosion of carbon steel tanks adequately addressed in the VCSNS LRA?

RAI B.2.1-3
The description of this program states that the inspection of above ground tanks will be consistent with the GALL Program XI.M32, "One-Time Inspection." However, in comparing this program with the one-time inspection program defined in GALL, the staff requests the applicant to address the following:

- Element 4, Detection of Aging Effects: This AMP does not discuss the qualification of the personnel conducting the inspection
- Element 6, Acceptance Criteria: This AMP does not discuss refer to the design minimum wall thickness nor to the criteria for verifying the absence of cracking.

B.2.10 Buried Piping and Tanks Inspection

RAI B.2.10-1
The buried piping and tanks inspection states in LRA Section B.2.10 under element 2 ("Preventive Actions") that underground components are coated and wrapped during installation to prevent them from directly contacting the soil environment. Briefly describe coating techniques and verify the adequacy of these techniques to provide isolation from the aggressive soil environment in light of the inadequacy of the cathodic protection system discussed in the operating history section.

RAI B.2.10-2
The buried piping and tanks inspection states in LRA Section B.2.10 under element 3 ("Parameters Monitored or Inspected") that the condition of coatings and wrappings will be determined by visual inspection whenever buried components are excavated for maintenance or for other reasons. The LRA later cited operating experience with buried piping and tanks that uses ultrasonic inspection technique. Will ultrasonic inspection techniques be used in addition to or in place of visual inspection? Provide justification.

RAI B.2.10-3
The buried piping and tanks inspection states in LRA Section B.2.10 under element 4 ("Detection of Aging Effects") that a specific inspection frequency for buried components is not warranted. Justify why periodic inspection of the most susceptible locations is not needed especially in areas with highest likelihood of corrosion problems and in areas with a history of corrosion problems.

RAI B.2.10-4
The buried piping and tanks inspection states in LRA Section B.2.10 under element 6 ("Acceptance Criteria") that the acceptance criteria are "no unacceptable degradation of coatings and wrappings that could result in loss of material and therefore a loss of component intended function, as determined by engineering evaluation." Will the coating and wrapping degradation be reported and evaluated according to site corrective actions procedure?

RAI B.2.10-5
Section B.2.10.1 discusses the operating experience with buried pipes and tanks. The applicant discusses the inspection of fuel oil storage tanks and associated piping performed as a result of the inadequacy of the cathodic protection system for these components. The staff

requests the applicant to discuss the operating experience/inspection of the other storage tanks and piping within the scope of this system.

V. C. Summer Request For Additional Information (RAI)

RAI Input for Summer License Renewal Application - by D. Shum

Section 2.3.3.5 Circulating Water System

RAI 2.3.3.5-1
Section 2.3.3.5 states that the only license renewal function of the circulating water system is to provide level instrumentation that will trip the circulating water pumps on high level to protect the inter-mediate and control buildings from flooding. There are no mechanical components/component types required for the circulating water system to perform its system intended function; therefore, no aging management review is required.

Since level instruments are not highlighted as within license renewal evaluation boundaries on flow diagrams, no flow diagrams have been marked for the circulating water system. Please clarify whether these level instruments are subject to an AMR, or justify their exclusion.

Section 2.3.3.7 Diesel Generators Services Systems

RAI 2.3.3.7-1
With regard to the diesel generator (DG) fuel oil storage and transfer system, the following components, (the vent line with flame arrestor for each fuel oil storage tank and each day tank, the manway for each fuel oil storage tank, and the fuel oil fill lines) are neither identified in drawing D-302-351 as being within the scope of license renewal nor included in LRA Table 2.3-23, which lists components subject to an AMR. The staff believes that these components are long-lived components with a passive function, and therefore are subject to an AMR. Please clarify whether these components are subject to an AMR, or justify their exclusion.

RAI 2.3.3.7-2
The components of DG crankcase vacuum system (e.g. crankcase pump cases, oil separators, flex connectors, valves, piping, etc.) are neither identified in drawings D-302-353 and IMS-32-005, Sheet 7 as being within the scope of license renewal nor included in LRA Table 2.3-23, which lists components subject to an AMR. The staff believes that these components are long-lived components with a passive function, and therefore are subject to an AMR. Please clarify whether these components are subject to an AMR, or justify their exclusion.

RAI 2.3.3.7-3
The components (e.g. expansion tanks, sight glasses, flex connectors, valves, piping, etc.) of DG jaket water system are neither identified in drawings D-302-353 and IMS-32-005, Sheet 4, as being within the scope of license renewal nor included in LRA Table 2.3-23, which lists components subject to an AMR. The staff believes that these components are long-lived components with a passive function, and therefore are subject to an AMR. Please clarify whether these components are subject to an AMR, or justify their exclusion.

V. C. SUMMER LRA
PLANT SYSTEMS BRANCH
DRAFT RAI'S FOR LRA SECTIONS 2.3.4.7 -12

By Bill Lefave

410.XX

Table 3.4-1, "Summary Of Aging Management Programs For The Steam & Power Conversion Systems Evaluated in NUREG- 1801 That Are Relied On For License Renewal," provides summaries for 13 AMR items. Of these 13 items, Items 1, 3, 8, 9, and 10, are not referenced in any of steam and power conversion subsections of Section 2.3.4, "Steam And Power Conversion Systems" of the LRA. While it appears that Items 3, 9, and 10, may not be applicable to the V. C. Summer plant, please explain why there are no references to Items 1 and 8 which appear to be applicable to the V. C. Summer plant. Also, revise the applicable LRA sections to add these references if necessary.

410.XX

Request for Additional Information: By Vince Klco

2.3.3.1-1 Ventilation damper housings are highlighted on the ventilation flow diagrams identified in the license renewal application (LRA) as within scope of license renewal. While ventilation damper housings are highlighted as within the scope of license renewal, ventilation damper housings are not identified in Table 2.3-18 of the application that relates ventilation system components types subject to an aging management review and their intended functions. Examples of ventilation damper housings highlighted on system flow diagrams include the following:

- Fuel Handling Building Charcoal flow diagram D-912-131, (B5, B3, D5, D3, E5, E8, F8).
- Reactor Building Cooling System flow diagram D-912-102 (C6, C10, D6, D9, G8).
- Auxiliary Building HEPA Exhaust System flows diagram D-912-120 (C8 and C10).
- Auxiliary Building Pump Room Cooling System flow diagram D-912-132 (H6, H7, J8)
- Control Room Normal and Emergency Air Handling System flow diagram D-912-140 (A1,B3,A6, B7, H5, H6, J7, H8, K7, H9, G12, A13)

State whether these components are within the scope of license renewal and subject to an aging management review (AMR). If so, provide the relevant information about the components in order to provide the staff with the ability to coordinate between the component/commodity tables and the flow diagram drawings, and complete the aging management review Table 2.3-18 of the LRA. If the components are not in scope or subject to an AMR, provide justification for their exclusion.

2.3.3.1-2 The following five passive components associated with ventilation system ductwork are not identified as within the scope of license renewal or subject to an aging management program:

- Ductwork turning vanes
- Ventilation system elastomer seals
- A ventilation equipment vibration isolator flexible connections
- Ductwork test connections
- Ductwork access doors

State whether you agree if these components are within the scope of license renewal and subject to an AMR. If they are, provide the information necessary to complete the aging management review result tables. If these components are not in scope and subject to an AMR, provide justification for their exclusion.

2.3.3.1-3 Clarify whether structural sealants used to maintain the power block building pressure boundary envelope (i.e., main control room, auxiliary building, fuel handling building, containment) at design pressure with respect to the adjacent areas are included in the scope of license renewal and subject to an aging management review. Provide information relating to structural sealants use as referenced in Table 2.1-3 on page 2.1-15 of NUREG-1800 (Standard Review Plan-License Renewal). The Standard Review Plan states that an applicant’s structural aging management program is expected to address structural sealants with respect to an AMR program. If structural sealants are not in the scope of license renewal and subject to an AMR, provide justification for their exclusion.

2.3.3.1-4 Filter housings in the air handling and local ventilation and cooling systems are identified on ventilation system flow diagrams referenced in the LRA as within the scope of license renewal. Filter housings perform the intended function of a pressure boundary. However, they are not included in the aging management review results Table 2.3-18 of the LRA. State whether filter housings are subject to an AMR and provide the relevant information about this component to enable the staff to complete its review of the aging management review result Table 2.3-18 in the LRA. If filter housings are not subject to an AMR, provide justification for their exclusion.

2.3.3.1-5 The safe shutdown controls and panels are not identified in section 9.4 of the FSAR. The Summer ventilation systems used to support use of the safe shutdown controls have not been included as part of the scoping and screening process. State whether the ventilation systems used to support the safe shutdown controls are within the scope license renewal and subject to an AMR in accordance with 10CFR54.4(a)(1) and (a)(2). If so, provide the relevant information about the components to enable the staff to complete its review of the aging management review result tables in the LRA. If the ventilation systems used to support the safe shutdown controls are not in the scope of license renewal and subject to an AMR, provide justification for their exclusion.

2.3.3-6 The air handling and local ventilation and cooling systems system scoping flow diagrams have highlighted instruments and their associated housings and tubing, indicating they are included in the scope of license renewal. State whether these identified instrument housings and their associated tubing are subject to an AMR and provide the relevant information within Table 2.3-18 to enable the staff to complete the license renewal review process. If the highlighted instrument housings and associated tubing are not subject to an AMR, provide justification for their exclusion.

2.3.3.1-7 The applicant does not describe their process of evaluating consumables in the license renewal application. The applicant should state whether their evaluation process for consumables is subject to screening guidance in accordance with Table 2.1-3 of NUREG-1800. If consumables are not considered subject to NUREG-1800 scoping and screening guidance, provide a justification for their exclusion.

Reviewer: H. A. Wagage

2.3.3.10 Gaseous Waste Processing System

RAI 2.3.3.10-1
The system flow diagram drawing, E-302-745, rev. 3 (catalytic hydrogen recombiner B) shows the piping of cooler condenser continue to drawing E-302-743. However, drawing E-302-743 is not included in the submittal, or referenced in Section 2.3.3.10 of the LRA. Explain whether the license renewal boundary of gaseous waste processing system extend to drawing E-302-743. Please supply this drawing

RAI 2.3.3.10-2
The system flow diagram drawing, E-302-742, rev. 11 (waste processing) does not identify the heat-exchanger-shell-chemical-drain piping and valve 7938A to be within the scope of license renewal. This piping and the housing of the valve provide a pressure retaining function and are passive and long-lived. Therefore, these components appear to be within the scope of license renewal and subject to an aging management review. Justify exclusion of these components form the scope of license renewal and aging management review.

RAI 2.3.3.10-3
The system flow diagram drawing, E-302-742, rev. 11 (waste processing) shows that waste gas compressor package numbers 1 and 2 are within the scope of licence renewal but not all components subject to an AMR. Detailed drawings of these packages are needed to review the components that are subject to an aging management review. Supply detail drawings that show these components and piping connections.

2.3.3.14 Liquid Waste Processing System

RAI 2.3.3.14-1
Section 2.3.3.14 of the LRA states that the licence renewal boundaries for the liquid waste processing system are depicted in drawing E-302-735. Table 2.3-28.of LRA lists components of "condensers" and "heat exchangers" subject to an AMR. However, drawing E-302-735 has identified only one heat exchanger, reactor coolant drain heat exchanger. Where can one locate the other heat exchanger/s and condensers in the LRA?

2.3.4.6 Gland Sealing Steam System

RAI 2.3.4.6-1
The gland seal system license renewal boundary drawing, D-302-141, rev. 15, does not identify the housings of stop valve, S.V. # 1. This valve provides a pressure retaining function. It appears that the housing of this valve is passive and long-lived and, as such, should be within the scope of license renewal and subject to an aging management review. Justify why this component is considered to be outside the scope of license renewal or is not subject to an aging management review.

DRAFT RAIs - SUMMER LRA (RAIs By Steve Jones)

2.3.3.6 Component Cooling Water

RAI 2.3.3.6-1
The following component types are shown to be within the scope of license renewal on the listed license renewal boundary drawings:

venturi Drawing D-302-614, locations D4, D5, D6, and D
radiation monitor housing Drawing D-806-005, locations A5 and A8

However, the staff is unable to locate these component types in Table 2.3-22 of the LRA. Clarify whether these component types are included in a component group already listed in the table. If not, justify the exclusion of these component types from being subject to an AMR in accordance with the requirements of 10 CFR 54.4(a) and 10 CFR 54.21(a)(1).

RAI 2.3.3.6-2
The license renewal boundary drawings referenced in Section 2.3.3.6 of the LRA show numerous lines connecting temperature elements or temperature indicators to piping segments identified as within license renewal scope. However, these connecting lines are identified as outside license renewal scope although they often include dimensional markings indicating they represent piping stubs. Describe the typical configuration used to monitor flow stream temperature in the component cooling water system using temperature elements or temperature indicators, and clarify which portions of these assemblies are subject to an AMR in accordance with the requirements of 10 CFR 54.4(a) and 10 CFR 54.21(a)(1).

2.3.3.9 Fuel Handling System (Jones)

RAI 2.3.3.9-1
The following components are shown to be within the scope of license renewal on license renewal boundary drawing D-302-651:

fuel transfer tube
- fuel transfer tube blank flange
- mechanical fasteners for blank flange
- valve body for fuel transfer tube gate valve
- piping and valve body for vent line connected to fuel transfer tube

However, Table 2.3-25 of the LRA lists only the fuel transfer tube as a component subject to an AMR. The fuel transfer tube and associated components perform a pressure boundary intended function for both containment integrity and spent fuel pool leakage prevention. Clarify whether each of the other components are included with the fuel transfer tube listed in the table. If not, add the components to Table 2.3-25 of the LRA or justify their exclusion from being subject to an AMR in accordance with the requirements of 10 CFR 54.4(a) and 10 CFR 54.21(a)(1).

NOTE FOR AGING MANAGEMENT REVIEW: Identified environments for these components fail to include stainless steel exposure to a borated water environment.

2.3.3.12 Instrument Air Supply System (Jones)

RAI 2.3.3.12-1
Section 9.3.1.3 of the Summer FSAR identifies the feedwater isolation valves as valves that are required to function following an accident and that do not fail in a safe position after a loss of air supply. These air operated valves are equipped with safety-related air accumulators to allow operation of the valves following a loss of air supply from the instrument air system. However, the staff did not identify these accumulators and, with the exception of the valve air operators, related components necessary for operation of the feedwater isolation valves among components identified as within the scope of license renewal in drawings referenced in

Sections 2.3.3.12 and 2.3.4.5 of the LRA. The air operators for the feedwater isolation valves were identified as within the scope of license renewal on license renewal drawing 1MS-25-898. Clarify whether the accumulators and related components necessary for the operation of the feedwater isolation valves are within the scope of license renewal and subject to an AMR. If not, justify their exclusion from being subject to an AMR in accordance with the requirements of 10 CFR 54.4(a) and 10 CFR 54.21(a)(1).

RAI 2.3.3.12-2
License renewal drawings referenced in Section 2.3.3.12 of the LRA identified the air accumulators and associated air components for the following valves and dampers as within the scope of license renewal:

control room outside air intake isolation dampers
service water makeup to component cooling water system isolation valves
emergency feedwater flow control valves
turbine-driven emergency feedwater pump steam isolation valve
main steam isolation valves
emergency diesel generator service water bypass valves
pressurizer power operated relief valves

However, the associated actuator was not identified as within the scope of license renewal. Clarify whether the portions of the actuators for the listed dampers and valves that perform a passive pressure boundary intended function are within the scope of license renewal and subject to an AMR. If not, justify their exclusion from being subject to an AMR in accordance with the requirements of 10 CFR 54.4(a) and 10 CFR 54.21(a)(1).

RAI 2.3.3.12-3
Section 9.2.1.2 of the FSAR states that the fire protection system serves as a standby means of cooling the diesel generators, and the cross-connect valve automatically opens on high lube oil temperature or high jacket water temperature when the diesel is operating in the emergency mode. Section 9.3.1.3 of the FSAR states that these fire protection system valves are equipped with quality related air accumulators. Describe the basis for excluding these air accumulators and associated air components from the scope of license renewal when the fire protection and service water system piping that interfaces at the valves is within the scope of license renewal.

2.3.3.21 Service Water System (Jones)

RAI 2.3.3.21-1
The license renewal boundary drawings referenced in Section 2.3.3.21 of the LRA show numerous lines connecting temperature elements or temperature indicators to piping segments identified as within license renewal scope. However, these connecting lines are identified as outside license renewal scope although they often include dimensional markings indicating they represent piping stubs. Describe the typical configuration used to monitor flow stream temperature in the service water system using temperature elements or temperature indicators, and clarify which portions of these assemblies are subject to an AMR in accordance with the requirements of 10 CFR 54.4(a) and 10 CFR 54.21(a)(1).

RAI 2.3.3.21-2
License renewal boundary drawing D-302-222 shows that service water piping extends to drawing D-302-085 at locations D12 and H12 for backup supply to the emergency feedwater pump suction and to drawing D-302-611 at locations B8 and G8 for supply of component cooling water system makeup water. However, Section 2.3.3.21 of the LRA fails to reference drawings D-302-085 and D-302-611 to include service water piping on these diagrams within the AMR for the service water system, and Tables 2.3-22 and 2.3-40 of the LRA, which present aging management results for the component cooling water and emergency feedwater systems respectively, do not reference aging management programs consistent with component exposure to a raw water environment. A related issue exists with regard to fire protection system piping that extends onto service water system drawing D-302-222 at locations B8-9 and J8-9 for supply of backup cooling water to the emergency diesel generators from the fire protection water system. Clarify how these piping segments have been included in an AMR and what aging management programs apply to these piping segments.

2.3.3.22 Spent Fuel Cooling System (Jones)

RAI 2.3.3.22-1
License renewal boundary drawing D-302-651 referenced in Section 2.3.3.22 of the LRA shows lines connecting temperature indicators to piping segments identified as within license renewal scope at locations D7, D6, F7, and F6. However, these connecting lines are identified as outside license renewal scope although they may represent piping stubs. Describe the typical configuration used to monitor flow stream temperature in the spent fuel cooling system using temperature indicators, and clarify which portions of these assemblies are subject to an AMR in accordance with the requirements of 10 CFR 54.4(a) and 10 CFR 54.21(a)(1).

V. C. Summer Request For Additional Information (RAI)

RAI Input for Summer License Renewal Application - by D. Shum

Section 2.3.3.5 Circulating Water System

RAI 2.3.3.5-1
Section 2.3.3.5 states that the only license renewal function of the circulating water system is to provide level instrumentation that will trip the circulating water pumps on high level to protect the inter-mediate and control buildings from flooding. There are no mechanical components/component types required for the circulating water system to perform its system intended function; therefore, no aging management review is required.

Since level instruments are not highlighted as within license renewal evaluation boundaries on flow diagrams, no flow diagrams have been marked for the circulating water system. Please clarify whether these level instruments are subject to an AMR, or justify their exclusion.

Section 2.3.3.7 Diesel Generators Services Systems

RAI 2.3.3.7-1
With regard to the diesel generator (DG) fuel oil storage and transfer system, the following components, (the vent line with flame arrestor for each fuel oil storage tank and each day tank, the manway for each fuel oil storage tank, and the fuel oil fill lines) are neither identified in drawing D-302-351 as being within the scope of license renewal nor included in LRA Table 2.3-23, which lists components subject to an AMR. The staff believes that these components are long-lived components with a passive function, and therefore are subject to an AMR. Please clarify whether these components are subject to an AMR, or justify their exclusion.

RAI 2.3.3.7-2
The components of DG crankcase vacuum system (e.g. crankcase pump cases, oil separators, flex connectors, valves, piping, etc.) are neither identified in drawings D-302-353 and IMS-32-005, Sheet 7 as being within the scope of license renewal nor included in LRA Table 2.3-23, which lists components subject to an AMR. The staff believes that these components are long-lived components with a passive function, and therefore are subject to an AMR. Please clarify whether these components are subject to an AMR, or justify their exclusion.

RAI 2.3.3.7-3
The components (e.g. expansion tanks, sight glasses, flex connectors, valves, piping, etc.) of DG jaket water system are neither identified in drawings D-302-353 and IMS-32-005, Sheet 4, as being within the scope of license renewal nor included in LRA Table 2.3-23, which lists components subject to an AMR. The staff believes that these components are long-lived components with a passive function, and therefore are subject to an AMR. Please clarify whether these components are subject to an AMR, or justify their exclusion.

V. C. SUMMER LRA
PLANT SYSTEMS BRANCH
DRAFT RAI'S FOR LRA SECTIONS 2.3.4.7 -12

By Bill Lefave

410.XX

Table 3.4-1, "Summary Of Aging Management Programs For The Steam & Power Conversion Systems Evaluated in NUREG- 1801 That Are Relied On For License Renewal," provides summaries for 13 AMR items. Of these 13 items, Items 1, 3, 8, 9, and 10, are not referenced in any of steam and power conversion subsections of Section 2.3.4, "Steam And Power Conversion Systems" of the LRA. While it appears that Items 3, 9, and 10, may not be applicable to the V. C. Summer plant, please explain why there are no references to Items 3 and 8 which appear to be applicable to the V. C. Summer plant. Also, revise the applicable LRA sections to add these references if necessary.

410.XX

Request for Additional Information: By Vince Klco

License Renewal Application
Plant Systems Branch
Division of Systems Safety and Analysis
Office of Nuclear Reactor Regulation
South Carolina Electric and Gas Company
V.C. Summer Nuclear Station
Docket Number 50-395

2.3.3.1-1 Ventilation damper housings are highlighted on the ventilation flow diagrams identified in the license renewal application (LRA) as within scope of license renewal. While ventilation damper housings are highlighted as within the scope of license renewal, ventilation damper housings are not identified in Table 2.3-18 of the application that relates ventilation system components types subject to an aging management review and their intended functions. Examples of ventilation damper housings highlighted on system flow diagrams include the following:

- Fuel Handling Building Charcoal flow diagram D-912-131, (B5, B3, D5, D3, E5, E8, F8).
- Reactor Building Cooling System flow diagram D-912-102 (C6, C10, D6, D9, G8).
- Auxiliary Building HEPA Exhaust System flows diagram D-912-120 (C8 and C10).
- Auxiliary Building Pump Room Cooling System flow diagram D-912-132 (H6, H7, J8)
- Control Room Normal and Emergency Air Handling System flow diagram D-912-140 (A1,B3,A6, B7, H5, H6, J7, H8, K7, H9, G12, A13)

State whether these components are within the scope of license renewal and subject to an aging management review (AMR). If so, provide the relevant information about the components in order to provide the staff with the ability to coordinate between the component/commodity tables and the flow diagram drawings, and complete the aging management review Table 2.3-18 of the LRA. If the components are not in scope or subject to an AMR, provide justification for their exclusion.

2.3.3.1-2 The following five passive components associated with ventilation system ductwork are not identified as within the scope of license renewal or subject to an aging management program:

- Ductwork turning vanes
- Ventilation system elastomer seals
- A ventilation equipment vibration isolator flexible connections
- Ductwork test connections
- Ductwork access doors

State whether you agree if these components are within the scope of license renewal and subject to an AMR. If they are, provide the information necessary to complete the aging management review result tables. If these components are not in scope and subject to an AMR, provide justification for their exclusion.

2.3.3.1-3 Clarify whether structural sealants used to maintain the power block building pressure boundary envelope (i.e., main control room, auxiliary building, fuel handling building, containment) at design pressure with respect to the adjacent areas are included in the scope of license renewal and subject to an aging management review. Provide information relating to structural sealants use as referenced in Table 2.1-3 on page 2.1-15 of NUREG-1800 (Standard Review Plan-License Renewal). The Standard Review Plan states that an applicant's structural aging management program is expected to address structural sealants with respect to an AMR program. If structural sealants are not in the scope of license renewal and subject to an AMR, provide justification for their exclusion.

2.3.3.1-4 Filter housings in the air handling and local ventilation and cooling systems are identified on ventilation system flow diagrams referenced in the LRA as within the scope of license renewal. Filter housings perform the intended function of a pressure boundary. However, they are not included in the aging management review results Table 2.3-18 of the LRA. State whether filter housings are subject to an AMR and provide the relevant information about this component to enable the staff to complete its review of the aging management review result Table 2.3-18 in the LRA. If filter housings are not subject to an AMR, provide justification for their exclusion.

2.3.3.1-5 The safe shutdown controls and panels are not identified in section 9.4 of the FSAR. The Summer ventilation systems used to support use of the safe shutdown controls have not been included as part of the scoping and screening process. State whether the ventilation systems used to support the safe shutdown controls are within the scope license renewal and subject to an AMR in accordance with 10CFR54.4(a)(1) and (a)(2). If so, provide the relevant information about the components to enable the staff to complete its review of the aging management review result tables in the LRA. If the ventilation systems used to support the safe shutdown controls are not in the scope of license renewal and subject to an AMR, provide justification for their exclusion.

2.3.3-6 The air handling and local ventilation and cooling systems system scoping flow diagrams have highlighted instruments and their associated housings and tubing, indicating they are included in the scope of license renewal. State whether these identified instrument housings and their associated tubing are subject to an AMR and provide the relevant information within Table 2.3-18 to enable the staff to complete the license renewal review process. If the highlighted instrument housings and associated tubing are not subject to an AMR, provide justification for their exclusion.

2.3.3.1-7 The applicant does not describe their process of evaluating consumables in the license renewal application. The applicant should state whether their evaluation process for consumables is subject to screening guidance in accordance with Table 2.1-3 of NUREG-1800. If consumables are not considered subject to NUREG-1800 scoping and screening guidance, provide a justification for their exclusion.

Virgil C. Summer License Renewal Application: Draft RAI
Reviewer: H. A. Wagage
Date: 12/6/02

2.3.3.10 Gaseous Waste Processing System

RAI 2.3.3.10-1
The system flow diagram drawing, E-302-745, rev. 3 (catalytic hydrogen recombiner B) shows the piping of cooler condenser continue to drawing E-302-743. However, drawing E-302-743 is not included in the submittal, or referenced in Section 2.3.3.10 of the LRA. Explain whether the license renewal boundary of gaseous waste processing system extend to drawing E-302-743. Please supply this drawing

RAI 2.3.3.10-2
The system flow diagram drawing, E-302-742, rev. 11 (waste processing) does not identify the heat-exchanger-shell-chemical-drain piping and valve 7938A to be within the scope of license renewal. This piping and the housing of the valve provide a pressure retaining function and are passive and long-lived. Therefore, these components appear to be within the scope of license renewal and subject to an aging management review. Justify exclusion of these components form the scope of license renewal and aging management review.

RAI 2.3.3.10-3
The system flow diagram drawing, E-302-742, rev. 11 (waste processing) shows that waste gas compressor package numbers 1 and 2 are within the scope of licence renewal but not all components subject to an AMR. Detailed drawings of these packages are needed to review the components that are subject to an aging management review. Supply detail drawings that show these components and piping connections.

2.3.3.14 Liquid Waste Processing System

RAI 2.3.3.14-1
Section 2.3.3.14 of the LRA states that the licence renewal boundaries for the liquid waste processing system are depicted in drawing E-302-735. Table 2.3-28.of LRA lists components of "condensers" and "heat exchangers" subject to an AMR. However, drawing E-302-735 has identified only one heat exchanger, reactor coolant drain heat exchanger. Where can one locate the other heat exchanger/s and condensers in the LRA?

2.3.4.6 Gland Sealing Steam System

RAI 2.3.4.6-1
The gland seal system license renewal boundary drawing, D-302-141, rev. 15, does not identify the housings of stop valve, S.V. # 1. This valve provides a pressure retaining function. It appears that the housing of this valve is passive and long-lived and, as such, should be within the scope of license renewal and subject to an aging management review. Justify why this component is considered to be outside the scope of license renewal or is not subject to an aging management review.

DRAFT RAIs - SUMMER LRA (RAIs By Steve Jones)

2.3.3.6 Component Cooling Water

RAI 2.3.3.6-1

The following component types are shown to be within the scope of license renewal on the listed license renewal boundary drawings:

venturi	Drawing D-302-614, locations D4, D5, D6, and D
radiation monitor housing	Drawing D-806-005, locations A5 and A8

However, the staff is unable to locate these component types in Table 2.3-22 of the LRA. Clarify whether these component types are included in a component group already listed in the table. If not, justify the exclusion of these component types from being subject to an AMR in accordance with the requirements of 10 CFR 54.4(a) and 10 CFR 54.21(a)(1).

RAI 2.3.3.6-2

The license renewal boundary drawings referenced in Section 2.3.3.6 of the LRA show numerous lines connecting temperature elements or temperature indicators to piping segments identified as within license renewal scope. However, these connecting lines are identified as outside license renewal scope although they often include dimensional markings indicating they represent piping stubs. Describe the typical configuration used to monitor flow stream temperature in the component cooling water system using temperature elements or temperature indicators, and clarify which portions of these assemblies are subject to an AMR in accordance with the requirements of 10 CFR 54.4(a) and 10 CFR 54.21(a)(1).

2.3.3.9 Fuel Handling System (Jones)

RAI 2.3.3.9-1

The following components are shown to be within the scope of license renewal on license renewal boundary drawing D-302-651:

fuel transfer tube
- fuel transfer tube blank flange
- mechanical fasteners for blank flange
- valve body for fuel transfer tube gate valve
- piping and valve body for vent line connected to fuel transfer tube

However, Table 2.3-25 of the LRA lists only the fuel transfer tube as a component subject to an AMR. The fuel transfer tube and associated components perform a pressure boundary intended function for both containment integrity and spent fuel pool leakage prevention. Clarify whether each of the other components are included with the fuel transfer tube listed in the table. If not, add the components to Table 2.3-25 of the LRA or justify their exclusion from being subject to an AMR in accordance with the requirements of 10 CFR 54.4(a) and 10 CFR 54.21(a)(1).

NOTE FOR AGING MANAGEMENT REVIEW: Identified environments for these components fail to include stainless steel exposure to a borated water environment.

2.3.3.12 Instrument Air Supply System (Jones)

RAI 2.3.3.12-1
Section 9.3.1.3 of the Summer FSAR identifies the feedwater isolation valves as valves that are required to function following an accident and that do not fail in a safe position after a loss of air supply. These air operated valves are equipped with safety-related air accumulators to allow operation of the valves following a loss of air supply from the instrument air system. However, the staff did not identify these accumulators and, with the exception of the valve air operators, related components necessary for operation of the feedwater isolation valves among components identified as within the scope of license renewal in drawings referenced in Sections 2.3.3.12 and 2.3.4.5 of the LRA. The air operators for the feedwater isolation valves were identified as within the scope of license renewal on license renewal drawing 1MS-25-898. Clarify whether the accumulators and related components necessary for the operation of the feedwater isolation valves are within the scope of license renewal and subject to an AMR. If not, justify their exclusion from being subject to an AMR in accordance with the requirements of 10 CFR 54.4(a) and 10 CFR 54.21(a)(1).

RAI 2.3.3.12-2
License renewal drawings referenced in Section 2.3.3.12 of the LRA identified the air accumulators and associated air components for the following valves and dampers as within the scope of license renewal:

control room outside air intake isolation dampers
service water makeup to component cooling water system isolation valves
emergency feedwater flow control valves
turbine-driven emergency feedwater pump steam isolation valve
main steam isolation valves
emergency diesel generator service water bypass valves
pressurizer power operated relief valves

However, the associated actuator was not identified as within the scope of license renewal. Clarify whether the portions of the actuators for the listed dampers and valves that perform a passive pressure boundary intended function are within the scope of license renewal and subject to an AMR. If not, justify their exclusion from being subject to an AMR in accordance with the requirements of 10 CFR 54.4(a) and 10 CFR 54.21(a)(1).

RAI 2.3.3.12-3
Section 9.2.1.2 of the FSAR states that the fire protection system serves as a standby means of cooling the diesel generators, and the cross-connect valve automatically opens on high lube oil temperature or high jacket water temperature when the diesel is operating in the emergency mode. Section 9.3.1.3 of the FSAR states that these fire protection system valves are equipped with quality related air accumulators. Describe the basis for excluding these air accumulators and associated air components from the scope of license renewal when the fire protection and service water system piping that interfaces at the valves is within the scope of license renewal.

2.3.3.21 **Service Water System** (Jones)

RAI 2.3.3.21-1
The license renewal boundary drawings referenced in Section 2.3.3.21 of the LRA show numerous lines connecting temperature elements or temperature indicators to piping segments identified as within license renewal scope. However, these connecting lines are identified as outside license renewal scope although they often include dimensional markings indicating they represent piping stubs. Describe the typical configuration used to monitor flow stream temperature in the service water system using temperature elements or temperature indicators, and clarify which portions of these assemblies are subject to an AMR in accordance with the requirements of 10 CFR 54.4(a) and 10 CFR 54.21(a)(1).

RAI 2.3.3.21-2
License renewal boundary drawing D-302-222 shows that service water piping extends to drawing D-302-085 at locations D12 and H12 for backup supply to the emergency feedwater pump suction and to drawing D-302-611 at locations B8 and G8 for supply of component cooling water system makeup water. However, Section 2.3.3.21 of the LRA fails to reference drawings D-302-085 and D-302-611 to include service water piping on these diagrams within the AMR for the service water system, and Tables 2.3-22 and 2.3-40 of the LRA, which present aging management results for the component cooling water and emergency feedwater systems respectively, do not reference aging management programs consistent with component exposure to a raw water environment. A related issue exists with regard to fire protection system piping that extends onto service water system drawing D-302-222 at locations B8-9 and J8-9 for supply of backup cooling water to the emergency diesel generators from the fire protection water system. Clarify how these piping segments have been included in an AMR and what aging management programs apply to these piping segments.

2.3.3.22 **Spent Fuel Cooling System** (Jones)

RAI 2.3.3.22-1
License renewal boundary drawing D-302-651 referenced in Section 2.3.3.22 of the LRA shows lines connecting temperature indicators to piping segments identified as within license renewal scope at locations D7, D6, F7, and F6. However, these connecting lines are identified as outside license renewal scope although they may represent piping stubs. Describe the typical configuration used to monitor flow stream temperature in the spent fuel cooling system using temperature indicators, and clarify which portions of these assemblies are subject to an AMR in accordance with the requirements of 10 CFR 54.4(a) and 10 CFR 54.21(a)(1).*Aging Management Evaluations in the GALL Report that Are Relied on for License Renewal, For Which GALL Recommends Further Evaluation*

3.1.2.2.2 Crack Initiation and Growth due to SCC

See RAI 3.1.2.4.2-3

3.1.2.2.3 Crack Initiation and Growth due to Stress Corrosion Cracking, Intergranular Stress Corrosion Cracking, and Thermal and Mechanical Loading

RAI 3.1.2.2.3-1
In LRA Appendix B.2.7, the applicant has identified the information sources that will be used to identify the susceptible locations in small-bore RCS piping and to select the sample locations for inspections. Confirm whether it will follow all on-going industry activities related to failure mechanisms for small-bore piping including the recommendations of the EPRI sponsored Materials Reliability Project (MRP) Industry Task Group (ITG) on Thermal Fatigue and will evaluate changes to inspection activities based on industry recommendations. Confirm also whether the samples locations selected for inspection will be the bounding locations for Class 1 small-bore piping within the scope of the license renewal.

3.1.2.2.4 Crack Initiation and Growth due to Stress Corrosion Cracking or Primary Water Stress Corrosion Cracking

RAI 3.1.2.2.4-1
In LRA Table 3.1-1, AMR Item 9, the applicant credits the Alloy 600 aging management program (LRA Appendix B.1.1) as one of the three programs for managing crack initiation and growth due to PWSCC of two Ni alloy components, core support pads and bottom head penetrations. In LRA Appendix B.1.1, the applicant also states that the Alloy 600 aging management program is consistent with GALL AMP XI.M11. However, according to Table 3.1-1 of NUREG-1800, the GALL AMP XI.M11 is credited for managing cracking only in CRD nozzles (i.e., vessel head penetrations) and no other Ni alloy components. Clarify this discrepancy.

RAI 3.1.2.2.4-2
The inservice inspection plan (LRA Appendix B.1.7) specifies ASME Section XI VT-3 examination to detect cracking of the core support pads. However, VT-3 examinations may not be sufficient to detect cracking of the core support pads. Submit an aging management program for managing cracking in core support pads and bottom head penetrations during the extended period of operation. Specifically, submit following information: (1) inspection method used in detecting cracking in these components, (2) technical basis showing adequacy of this method to detect cracking, (3) inspection frequency and its justification, and (4) acceptance criteria.

RAI 3.1.2.2.4-3
In LRA Table 3.1-1, AMR Item 11, the applicant credits LRA Appendix B.1.1, Alloy 600 aging management program, which is a condition-monitoring program, for managing cracking of Alloy 82/182 welds for the pressurizer instrumentation penetrations and heater sleeves due to PWSCC. However, in LRA Appendix B.1.1, the applicant states that this program is consistent with NUREG/CR-1801 AMP XI.M11, which includes only reactor pressure vessel head penetrations in its scope and no other Alloy 600 components. Therefore, the applicant needs

to modify the scope of its program (LRA Appendix B.1.1) to include all other Alloy 600 components in addition to reactor vessel head penetrations.

RAI 3.1.2.2.4-4
The applicant has not presented aging management review results for pressurizer heater sheaths. Confirm whether the heater sheaths at VCSNS are made of Alloy 600. If so, then provide a program for managing cracking of the heater sheaths due to PWSCC.

3.1.2.2.5 Loss of Fracture Toughness due to Neutron Irradiation Embrittlement

RAI 3.1.2.2.5-1
The reactor vessel internals program assumes sufficient redundancy in bolt function that the plant can continue to function safely with a fewer than 100% of the bolts intact. The staff finds that this approach is consistent with the one described in NUREG-1801, GALL Section XI.M16 ("PWR Vessel Internals"). However, the staff needs additional information for evaluating the program. With respect to the application of this program to the detection of irradiation embrittlement of the baffle former bolts, identify the neutron fluence threshold for which the baffle former bolts become susceptible to loss of fracture toughness due to neutron irradiation embrittlement and void swelling, submit technical justification for the threshold, identify the percentage of the bolts to be selected for inspection, and submit the technical basis for this selection process.

3.1.2.2.6 Crack Growth due to Cyclic Loading

RAI 3.1.2.2.6-1
In LRA Table 3.1-1, AMR Item 7, the applicant states that the VCSNS vessel is constructed of ASME SA 533 Grade B, Cl 1 plate material and not ASME SA 508 Cl 2 forgings. Therefore, the aging effect of growth of underclad crack is not applicable to the VCSNS vessel. However, Table 5.2-8 of the VCSNS FSAR identifies SA 508 Cl 2 as one of the materials for reactor vessel shell, flange and nozzle forgings and nozzle safe ends. Clarify this discrepancy.

3.1.2.2.7 Crack Initiation and Growth due to SCC and Irradiation-Assisted Stress Corrosion Cracking (IASCC)

RAI 3.1.2.2.7-1
Under the reactor vessel internals inspection program, the applicant has selected volumetric inspection as the plant-specific basis for addressing the issue of crack initiation and growth due to SCC and irradiation-assisted stress corrosion cracking in the baffle former bolts. However, the staff needs additional information for evaluating the program. With respect to the application of this program to the detection of IASCC cracking of the baffle former bolts, identify the neutron fluence threshold for which the baffle former bolts become susceptible to IASCC cracking, submit technical justification for the threshold, identify the percentage of the bolts to be selected for inspection, and submit the technical basis for this selection process.

Aging Management Programs (System Specific)

Small-Bore Class 1 Piping Inspection

RAI 3.1.2.3.1-1 This RAI is the same as RAI 3.1.2.2.3-1.

Alloy 600 Aging Management Program

RAI 3.1.2.3.2-1
The Alloy 600 management program (LRA Section B.1.1) relies on detecting PWSCC cracks in head penetrations by means of inspection for signs of boric acid leakage during outages. Submit the following additional information regarding the boric acid leakage inspection: (1) Confirm that the boric acid leakage inspection includes inspection of bare vessel head (insulation being removed from the vessel head prior to inspection), (2) confirm that after the inspection vessel head is cleaned of any boric acid deposits prior to installing the insulation, (3) confirm whether ASME VT-2 examination method is used to detect leakage through a crack in the vessel head penetration, and (4) since the leakage through a PWSCC crack is generally very small, provide technical basis ensuring that the boric acid leakage inspection will be able to detect such a small leakage.

RAI 3.1.2.3.2-2
In LRA Section B.1.1, the applicant states that PWSCC cracks can also be detected by monitoring primary coolant leakage per Technical Specifications during plant operation. It is unlikely that monitoring of primary coolant leakage would be sensitive enough to detect a very small leakage through a PWSCC crack. Submit operating experience supporting the use of primary coolant leakage monitoring during operation for detecting a PWSCC crack.

RAI 3.1.2.3.2-3
In response to NRC Bulletin 2001-01, the applicant states that, in 1999, VCSNS performed VT-3 inspections of the vessel head interior surface and did not find any recordable indications. Identify the objective of that inspection and confirm whether it can reliably detect any cracking or loss of material at the vessel head interior surface.

RAI 3.1.2.3.2-4
Confirm whether any plant-specific augmented inspection program for vessel head penetration, including a combination of surface and volumetric inspections, was developed in response to NRC GL 97-01.

RAI 3.1.2.3.2-5
NRC Bulletin 2002-2 was issued after the submittal of the LRA. Describe how the discussion in this bulletin on adequacy of the current inspection requirements and programs for vessel head penetrations would impact the VCSNS Alloy 600 aging management program.

RAI 3.1.2.3.2-6
As suggested in the NRC closure letter from K. R. Cotton to G. J. Taylor, dated December 17, 1999 for SCE&G response to Generic Letter 97-01; the LRA needs to include a summary of the results of any inspections that have been completed on VCSNS vessel head penetrations prior to the license renewal application. Therefore, submit the following information for these

inspections: (1) number of vessel head penetrations inspected and their locations on the vessel head, (2) inspection methods used, (3) number of Alloy 82/182 attachment welds inspected, and (4) inspection results.

RAI 3.1.2.3.2-7
The applicant states that the program will be enhanced according to the changes indicated by emerging regulatory requirements and identified by industry programs. However, the Alloy 600 aging management program, described in LRA Section B.1.0, does not specify whether the applicant would participate in the industry program for managing PWSCC type aging on vessel head penetrations. Confirm VCSNS participation in the industry program.

RAI 3.1.2.3.2-8
The FSAR supplement for the Alloy 600 management program (LRA Section B.1.1) is presented in LRA Appendix A, Section 18.2.4. The supplement states that the pressurizer and steam generator subcomponents in addition to the vessel subcomponents are within the scope of the program. However, because the applicant states that the program is consistent with the GALL AMP XI.M11, the scope of the program is limited to only vessel head penetrations and does not include other Alloy 600 components. Clarify this discrepancy and modify the supplement accordingly.

<u>Reactor Vessel Surveillance Program</u>

RAI 3.1.2.3.3-1
In accordance with ASTM E185, for current 40-year practice, it is recommended that the last capsule to be removed should receive the same or higher fluence than the peak EOL fluence. Therefore, the applicant should confirm that the updated capsule removal schedule reflects a capsule to be withdrawn with a predicted fluence equal to or greater than the peak EOL fluence for the extended period of operation for VC Summer.

RAI 3.1.2.3.3-2
Provide information regarding the fluence calculation methodology, i.e., is it consistent with the recommendations of DG-1053 and RG 1.190? In addition, confirm whether an alternative dosimetry will be used at VCSNS to monitor neutron fluence during the period of extended operation.

3.1.2.3.4 <u>Reactor Head Closure Studs Program</u>

RAI 3.1.2.3.4-1
In LRA Table 3.1-1, AMR Item 18, the applicant states that the aging effect requiring management is loss of closure integrity rather than cracking of closure studs. Describe the difference between these two aging effects and justify the selection of loss of closure integrity as an aging effect requiring management.

RAI 3.1.2.3.4-2
The applicant states that this program manages the effects of stress relaxation in the reactor head closure studs. Explain how this program manages the effects of stress relaxation.

3.1.2.3.5 Reactor Vessel Internals Inspection

RAI 3.1.2.3.5.2-1
In LRA Appendix B, Section B.2.4, the applicant describes its AMP to manage aging processes in reactor vessel internals. The LRA states that this AMP is consistent with GALL AMP XI.M16, with the clarification that the resolution criterion for the enhanced VT-1 examination at the Summer Plant is expected to be less than 0.0005-in. resolution, which is specified in the GALL program. Submit technical justification for using less than 0.0005-in resolution, and explain how the anticipated reduction in the resolution criterion will be determined.

RAI 3.1.2.3.5-2
LRA Section B.2.4 describes the VCSNS Reactor Vessel Internals Inspection Program, a new program to assess the condition of reactor vessel internals in order to assure that the applicable aging effects will not result in loss of the intended functions during the period of extended operation. With respect to the irradiation embrittlement of the RV internal components, the staff notes that NUREG/CR-6048, Oak Ridge National Laboratory, has used $5x10^{20}$ neutrons/cm^2 (E > 0.1 MeV) as the threshold for loss of fracture toughness due to radiation embrittlement in Type 304 austenitic stainless steel materials. Confirm whether this threshold value will be used at VCSNS for austenitic stainless steel vessel internals. If an alternate value is proposed, then submit a technical basis for that alternate value. Also provide the technical basis for the selection of the RV internal components for inspection. This RAI is similar to RAI 3.1.2.2.5-1.

RAI 3.1.2.3.5-3
LRA Section B.2.4 describes the VCSNS Reactor Vessel Internals Inspection Program, a new program to assess the condition of reactor vessel internals in order to assure that the applicable aging effects will not result in loss of the intended functions during the period of extended operation. With respect to the application of this program to the detection of irradiation-assisted stress corrosion cracking of RV internals components, the staff requests additional information on how the applicant determines which RV internal components are susceptible to irradiation-assisted stress corrosion cracking, what components will be selected for inspection, and what the technical basis is for this selection process. This RAI is similar to RAI 3.1.2.2.7-1.

RAI 3.1.2.3.5-4
LRA Section B.2.4 describes the VCSNS Reactor Vessel Internals Inspection Program, a new program to assess the condition of reactor vessel internals in order to assure that the applicable aging effects will not result in loss of the intended functions during the period of extended operation. The staff reviewed the applicant's FSAR supplement (LRA Section 18.2.18) to verify that it provides an adequate description of the programs credited with managing this aging effect, as required by 10 CFR 54.21(d). The staff notes that the description of the Reactor Vessel Internals Inspection in LRA Section B.2.4 states that "specific acceptance criteria for changes in dimension due to void swelling, loss of preload due to stress relaxation, and loss of material due to wear will be determined by analysis as part of the inspection plan." The staff requests that the applicant commit to supplement the reactor vessel internals inspection program and to submit an integrated report to thte NRC prior to the end of the initial operating term for VC Summer. The report should summarize the

understanding of the aging effects applicable to the reactor vessel internals and should contain a description of the VC Summer inspection plan, including methods for detection and sizing of cracks and acceptance criteria. This should also be discussed in the FSAR supplement

3.1.2.3.6 Bottom-Mounted Instrumentation Inspection

RAI 3.1.2.3.6-1
The applicant states that there is no NUREG-1801 (GALL report) item addressing the bottom-mounted instrumentation inspection program. This is not so. Item IV.B2.6-c of NUREG-1801 refers to the recommendations of NRC I&E Bulletin 88-09, “Thimble Tube Thinning in Westinghouse Reactors.” Confirm whether the bottom-mounted instrumentation inspection program follows all the recommendations of the bulletin. If not, then submit a justification for not following these recommendations.

RAI 3.1.2.3.6-2
The applicant states that the bottom-mounted instrumentation inspection program monitors tube wall degradation in 100% of the BMI thimble tubes using eddy current testing. Submit information about whether the entire length of each thimble tube is inspected or only a selected portion of the length and present corresponding technical basis.

RAI 3.1.2.3.6-3
The applicant states that the frequency of ECT examination is based on an analysis of data obtained using wear rate relationships that are predicted based on Westinghouse research. Submit an explanation for the wear rate relationship and describe the Westinghouse research mentioned here.

RAI 3.1.2.3.6-4

(a) The applicant states that the ECT results are trended, wear rates are calculated, and inspections are planned prior to the refueling outage in which the thimble tube wear is predicted to exceed the acceptance criteria. Regarding the predicted wear rate, NRC I&E Bulletin 88-09 states that, based on the available data, it is not possible to accurately predict thimble tube wear rates. Explain how this difficulty in accurately predicting thimble tube wear rates is taken into account in developing the applicant’s plan for the next thimble tube inspection.

(b) In describing its operating experience, the applicant states that the analysis of the wear rate data derived from the inspections performed at RF-4 and RF-5 determined that the next inspection of the thimble tubes is not required until RF-14. Explain and justify the use of this extrapolation of the limited inspection results for scheduling the next inspection of the thimble tubes.

RAI 3.1.2.3.6-5
The bottom-mounted instrumentation inspection program uses 75% loss of initial wall thickness as an acceptance criterion. Provide the technical justification for this criterion and explain how the allowances for such items as inspection methodology and wear scar geometry uncertainties, which were mentioned in NRC I&E Bulletin 88-09, are included in the criterion.

RAI 3.1.2.3.6-6

(a) The bottom-mounted instrumentation inspection program requires that the thimble tubes must be capped or repositioned if the projected through-wall wear exceeds 75% prior to the next scheduled ECT. Explain the factors that determine whether an affected thimble tube is to be capped or repositioned. If an affected tube is to be repositioned, then explain how the distance for repositioning is determined.

(b) The bottom-mounted instrumentation inspection program also requires that the thimble tube must be capped or replaced if projected through wall wear exceeds 80% prior to the next scheduled ECT. Explain the factors that determine whether an affected thimble tube is to be capped or replaced.

RAI 3.1.2.3.6-7
Since the issuance of IE Bulletin 88-09, the applicant has performed two inspections (RF-4 and RF-5) on thimble tubes at VCSNS. The applicant reports that several thimble tubes were repositioned in RF-5, but no thimble tubes were capped or required replacement. Confirm whether all the thimble tubes were inspected during these two inspections, and explain how it was determined to reposition several of these tubes.

Aging Management of Plant-Specific Components

3.1.2.4.1 <u>Reactor Coolant System Non-Class 1 Components</u>

RAI 3.1.2.4.1-1
LRA Table 3.1-2, AMR Item 14, states that stainless steel piping and piping system components including non-Class 1 pipe and valve bodies are internally exposed to treated water from the reactor makeup water system for pressurizer relief tank spray. As a result, these components are subject to loss of material due to crevice and pitting corrosion, and the chemistry program, LRA Appendix B.1.4, Chemistry Program is credited with managing this aging effect for these components. The applicant states that the chemistry program is similar to GALL AMP XI.M2, except it does not include inspection of selected components to verify the effectiveness of the program. The applicant further states that its evaluation of the applicable aging effect and the corresponding aging management program are consistent in material and environment, aging effect and credited program with the GALL Chapter VII.C2, “Closed Cycle Cooling Water System.” However, GALL AMP XI.21, “Closed Cycle Cooling Water System,” requires inspection to detect loss of material due to corrosion at locations of stagnant flow condition and crevices. Identify or submit an aging management program to verify the effectiveness of the chemistry program to manage loss of material.

RAI 3.1.2.4.1-2
According to LRA Table 3.3-1, AMR Item 14, the applicant credits the chemistry program (LRA Appendix B.1.4) for managing loss of material in RCP thermal barrier flange. The applicant states that the chemistry program is similar to GALL AMP XI.M21, except it does not include inspection. The use of the chemistry program alone may be inadequate because the program does not require inspection of the components to determine whether loss of material is taking place. The GALL AMP XI.M21 requires such inspection. Therefore, confirm that the

effectiveness of the chemistry program is verified for the non-Class 1 RCP thermal barrier flange and piping/tubing

3.1.2.4.2 Reactor Coolant Piping, Valves and Pumps

RAI 3.1.2.4.2-1
LRA Table 3.1-1, AMR Item 22, identifies loss of closure integrity rather than loss of preload and cracking as an aging effect for stainless steel and low-alloy steel bolting requiring management. Explain how the management of loss of closure integrity instead of loss of preload and cracking would ensure that the intended function of the bolted joint (pressure boundary integrity) would be maintained during the extended period of operation.

RAI 3.1.2.4.2-2
LRA Table 3.1-1, AMR Item 20, states that the CASS elbows and nozzles of the RCS Class 1 piping are not susceptible to loss of fracture toughness because these components have low molybdenum content and have delta ferrite levels of less than 20%. This is acceptable because the material chemistry for these components meets the screening criteria set forth in the letter dated March 19, 2000, from Christopher Grimes, NRC, to Douglas Walters, NEI. This AMR Item is, however, not identified in LRA Table 2.3-2. Clarify this discrepancy.

RAI 3.1.2.4.2-3
The austenitic stainless steel RCS piping is susceptible to stress corrosion cracking at their external surface if it comes in contact with halogens that may be present in the thermal insulation. Cracking has not been identified as an aging effect at the external surface of these components. Confirm that all insulation used on austenitic stainless steel RCS piping to ensure that the piping is not susceptible to stress-corrosion cracking from halogens. Note that this is identified as License Renewal Action Item 4 by the industry report WCAP-14575-A, “Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components, December 2000.”

RAI 3.1.2.4.2-4
LRA Table 3.1-1, AMR Item 22, identifies boric acid corrosion surveillances (LRA Appendix B.1.2) and the inservice inspection plan (LRA Appendix B.1.7) for managing aging effects in the RCS Class 1 bolted closures. Clarify how these two programs are sufficient to manage aging effects of loss of preload and cracking of RCS Class 1 bolting such that its intended function (pressure boundary) is maintained during the extended period of operation.

RAI 3.1.2.4.2-5
The NRC Information Notice 2000-17, “Crack in Weld Area of Reactor Coolant System Hot Leg Piping at V. C. Summer,” reports a crack in Alloy 182/82 weld between the ‘A’ hot leg nozzle and stainless steel piping. Submit the following information related to this event:

(a) Explain how this cracking event has been taken into account in the ISI Plan (LRA Appendix B.1.7)

(b) The operating experience is described in LRA Appendix B.1.1, Alloy 600 Aging Management Program but it is not clear whether this program is credited for managing PWSCC cracking in Alloy 82/182 welds in RCS Class 1 piping. Clarify this.

(c) Identify any mitigative actions (e.g., mechanical stress improvement) taken since the submittal of the LRA to minimize the growth of existing PWSCC cracks and describe any plan for ensuring the effectiveness of these actions during the extended period of operation.

RAI 3.1.2.4.2-6
The chemistry program (LRA Appendix B.1.4) references water quality that is compatible with the materials of construction used in the Class 1 piping and associated components in order to minimize loss of material and cracking. This program incorporates EPRI and Institute of Nuclear Power Operations (INPO) guidelines, which reflect industry experience, and the "lessons learned" from VCSNS and external industry operating experience. Confirm whether the chemistry program incorporates the guidelines in EPRI TR-105714 (Rev. 3 or later revisions or update). Identify any differences between the chemistry program and these guidelines and submit technical justification for these differences.

RAI 3.1.2.4.2-7
According to LRA Table 2.3-2, the results for austenitic stainless steel piping and fittings (less than NPS 4"), and orifices exposed to chemically treated borated coolant are presented in LRA Table 3.1-1, AMR Item 6, and LRA Table 3.1-2, AMR Item 6. Both AMR items identify the same aging effect (i.e., cracking) but different aging management programs. AMR Item 6 of LRA Table 3.1-1 credits three programs, the chemistry program (LRA Appendix B.1.4), the ISI plan (LRA Appendix B.1.7), and the small-bore Class 1 piping inspections (LRA Appendix B.2.7) for managing cracking. However, AMR Item 6 of LRA Table 3.1-2 credits only one program, the chemistry program (LRA Appendix B.1.4), for managing cracking. Explain this apparent discrepancy.

3.1.2.4.3 Reactor Vessel

RAI 3.1.2.4.3-1
LRA Table 2.3-3 refers to LRA Table 3.1-1, AMR Item 24, for AMR results for stainless steel cladding on reactor vessel closure head dome, closure head and vessel flanges, and bottom head. The AMR item identifies cracking as an aging effect requiring management. However, the GALL report, NUREG 1801, does not identify cracking as an aging effect for cladding on these components, which are made of SA 533B, Cl 1. Clarify this inconsistency. In addition, submit technical basis for identifying cracking as an applicable aging effect for the cladding.

RAI 3.1.2.4.3-2
In LRA Table 3.1-1, AMR Item 22, the applicant states that loss of material due to wear is not considered a valid aging effect for control rod drive flange bolting requiring management. This statement implies that VCSNS has installed control rod drive flange bolting. However, Section 5.4.2 of the VCSNS FSAR states that the upper ends of the CRD nozzles have a welded flexible canopy seal and not bolting. Explain this discrepancy.

RAI 3.1.2.4.3-3
LRA Table 2.3-3 refers to LRA Table 3.1-1, AMR Item 28, for the AMR results for the reactor vessel closure studs assembly. However, LRA Table 3.1-1, AMR Item 28, presents the AMR results for vessel and vessel closure head flanges and not for closure studs assembly. Explain this discrepancy. (Note that according to the GALL report, the component group addressed by the AMR Item 28 should have included reactor vessel and reactor vessel closure head flanges instead of reactor vessel closure studs.)

RAI 3.1.2.4.3-4
The austenitic stainless steel and Ni-alloy base reactor vessel appurtenances (i.e., CRD housings, vessel head penetrations, and Alloy 82/182 welds) are susceptible to stress corrosion cracking at the external surface if they come in contact with halogens that may be present in the thermal insulation. The applicant has not identified cracking as an aging effect at the external surface of these components. Submit a description of all insulation used on austenitic stainless steel and Ni-alloy base reactor vessel components and demonstrate that these components are not susceptible to stress-corrosion cracking from halogens.

RAI 3.1.2.4.3-7
The chemistry program (LRA Section B.1.4) also manages loss of material in stainless steel and Ni-alloy reactor vessel components (i.e., CRD housings, cladding, vent plug, bottom head and closure head penetration tubes, reactor vessel core support pads, and nozzle safe ends) internally exposed to chemically treated borated coolant. The staff does not believe that the chemistry program alone can effectively manage loss of material, especially due to crevice corrosion, in these components. The applicant needs to provide an AMP to ensure that no unacceptable loss of material is occurring in these components.

3.1.2.4.4 <u>Reactor Vessel Internals</u>

RAI 3.1.2.4.4-1
In LRA Table 3.1-1, AMR Items 5 and 31, the applicant identifies loss of fracture toughness due to irradiation as one of the applicable aging effects for the stainless steel reactor vessel internals in the fuel zone region. The applicant's identification of all of the reactor vessel internals in the fuel zone region as being susceptible to loss of fracture toughness due to irradiation represents an acceptable position to the staff. However, the staff needs additional information. Submit a criterion used to identify the vessel internals that are susceptible to loss of fracture toughness due to neutron irradiation along with its technical basis, and explain why the reactor vessel internals outside the fuel zone region are not considered susceptible to loss of fracture toughness due to irradiation.

RAI 3.1.2.4.4-2
The applicant credits the Reactor Vessels Internals Inspection Program (LRA Appendix B.2.4) alone with managing loss of preload due to stress relaxation in VCSNS hold-down spring, clevis insert bolts, and upper and lower support column bolts (LRA Table 3.1-1, AMR Items 30 and 35). In contrast, NUREG-1801 specifies that both inservice inspection and loose parts monitoring for managing loss of preload due to stress relaxation in the lower and upper support column bolts (GALL Items IV, B2.1-k and B2.5-h). For the hold-down spring (GALL Item B2.1-d) and clevis insert bolts (GALL Item IV, B2.5-i), NUREG-1801 states that either

loose parts monitoring or neutron noise monitoring is to be used in addition to inservice inspection to manage loss of preload. Explain how the Reactor Vessels Internals Inspection Program alone in the absence of either loose parts monitoring or neutron noise monitoring will adequately manage loss of preload in these components.

3.1.2.4.5 In-core Instrumentation System

3.1.2.4.5-1 In LRA Table 3.1-2, AMR Item 4 identifies stainless steel as material for in-core thermocouple seal bolting. However, in Discussion column for this AMR item, the applicant refers to high strength material for this bolting. Clarify this discrepancy. If high-strength, low-alloy steel is the bolting material, then explain why loss of material due to boric acid corrosion caused by leaking borated coolant is not an aging effect for this bolting material.

3.1.2.4.5-2 The applicant has identified loss of closure integrity rather than loss of preload and cracking as an applicable aging effect requiring management for closure bolting for in-core thermocouple seals. Explain how managing of loss of closure integrity would ensure that the pressure boundary of the bolted joint would be maintained during the extended period of operation.

3.1.2.4.5-3 In Westinghouse-designed PWRs, mechanical high-pressure seals, located at the seal table, are used to seal the area between the thimble tubes and the long-radius guides. Describe how the sealing of the area between the thimble tube and the guide is achieved at VCSNS and confirm whether bolted connection is employed for this mechanical seal. If a bolted connection is employed, then identify applicable aging effects and present an AMP for managing these effects.

3.1.2.4.5-4 The inservice inspection plan program (LRA Section B.1.7) is credited for managing loss of mechanical closure integrity, which includes loss of preload, loss of material and cracking of the bolted closures for the in-core thermocouple seal assemblies. The inservice inspection plan, which is based on ASME Section XI, Subsection IWB, requires VT-1 visual examination of bolts. Explain how the VT-1 examination can manage the effect of loss of preload so that the intended function of the bolted closure, i.e., pressure boundary, is maintained during the extended period of operation.

3.1.2.4.6 Pressurizer

RAI 3.1.2.4.6-1

The applicant states that the identification of the applicable aging effects for the pressurizer in LRA Table 3.1-1 is consistent with the GALL report. However, The GALL report presents an AMR for five additional pressurizer components [pressurizer seismic lugs, heater elements (heater sheaths), manway pad gasket seating surface, safety valves, and relief valves] that are not addressed in the LRA. According to Table 2-1 in the Westinghouse report WCAP 14574-A "License Renewal Evaluation: Aging Management Evaluation for Pressurizers," the first three components are within the scope of license renewal and require AMRs. Provide technical justification for not presenting an AMR for these three components, i.e., pressurizer seismic lugs, heater elements, and manway pad gasket seating surface. Also explain why the AMR results for safety and relief valves are not presented in LRA Table 3.1-1.

RAI 3.1.2.4.6-2
According to LRA Table 2.3-6, the applicant presented an AMR for pressurizer nozzles and safe ends. However, it is not clear to the staff about which specific nozzles are addressed by the LRA. Confirm whether the following five pressurizer nozzles and safe ends are included: surge nozzle, spray nozzle, safety nozzle, relief nozzle, and their safe ends, and instrument nozzle.

RAI 3.1.2.4.6-3
In LRA Table 2.3-6, the applicant presented AMR of manway cover (Row 4) and manway forgings (Row 7) exposed to chemically treated borated coolant. Why are the AMR results for these two components different? Does the AMR of manway forgings include that of manway flanges?

RAI 3.1.2.4.6-4
According to Section 3.2.5 of the Westinghouse report, WCAP-14574-A, "License Renewal Evaluation: Aging Management Evaluation for Pressurizers," four components of the pressurizer for which an AMR is performed, are exposed to fluid flows that have the potential to result in erosion of the components: surge nozzle thermal sleeve and safe end, and spray nozzle thermal sleeve and safe end. The applicant has not identified loss of material due to erosion as an applicable aging effect for these components. Explain why loss of material due to erosion is not an applicable aging effect for these components. If loss of material due to erosion is an applicable aging effect, then provide an AMP for managing it.

RAI 3.1.2.4.6-5
The attachment welds at the inside surface of the pressurizer are susceptible to cracking due to stress corrosion cracking if they are sensitized during fabrication. The applicant has not presented an AMR for these welds. Identify the components that are welded to the inside surface of the pressurizer and provide technical justification for determining whether cracking due to SCC is an applicable aging effect. If cracking is an applicable aging effect for the attachment welds, then provide an AMP for managing this effect.

RAI 3.1.2.4.6-6
In LRA Table 3.1-2, AMR Item 7, the applicant credits the chemistry program (LRA Appendix B.1.4) for managing loss of material due to crevice and pitting corrosion in the pressurizer shell and heads cladding with austenitic stainless steel and stainless steel components internally exposed to chemically treated borated coolant. These components are susceptible to crevice and pitting corrosion because high levels of oxygen may be present in the PWR reactor coolant. However, PWR licensees maintain hydrogen overpressure in the reactor coolant, and if the overpressure were at sufficiently high levels, it would provide protection in creviced geometries on the internal surfaces of the pressurizer. Explain how the chemistry program will provide for a sufficient level of hydrogen overpressure to manage crevice corrosion of the internal surfaces of the pressurizer.

RAI 3.1.2.4.6-7
LRA Table 3.1-1, AMR Item 24, credits the chemistry program (LRA Appendix B.1.4) and in-service inspection plan program (LRA Appendix B.1.7) for managing cracking of the pressurizer shell, lower head and upper head cladding with austenitic stainless steel and internally exposed to chemically treated borated coolant. The in-service inspection plan is

mainly directed at structural welds in the pressurizer shell and heads and not at stainless steel cladding. However, in 1990, crack-like indications were discovered in the Haddam Neck pressurizer cladding. Thermal fatigue can initiate and propagate such cracking through the cladding and into the ferritic base metal or weld metal beneath the clad. Therefore, submit an AMP to verify whether thermal fatigue-induced cracking has initiated in the clad and propagated through it into the ferritic base metal or weld metal beneath the clad.

RAI 3.1.2.4.6-8
LRA Table 3.1-1, AMR Item 26, credits the boric acid corrosion surveillance program (LRA Appendix B.1.2) for managing loss of material due to boric acid corrosion of pressurizer carbon steel and low alloy steel components: shell, upper and lower heads, nozzles, integral support, and manway cover and bolts. Provide detailed information showing how the program will be sufficient to manage the corrosive effects of boric acid leakage on these components during the extended period of operation, including postulated leakage from the pressurizer nozzle-to-vessel welds, pressurizer nozzle-to-safe end welds, and pressurizer manway bolting materials.

RAI 3.1.2.4.6-9
LRA Table 3.1-1, AMR Item 22, credits the in-service inspection plan program (LRA Appendix B.1.7) for managing loss of mechanical closure integrity, which includes loss of preload, loss of material and cracking, of the bolted closures for the pressurizer manway cover bolts. The inservice inspection plan, which is based on ASME Section XI, Subsection IWB, requires volumetric, and VT-1 and VT-2 visual examinations of bolts. Explain how these examinations manage the effects of loss of preload so that the intended function of the bolted closure, i.e., pressure boundary, is maintained during the extended operation.

3.3 <u>Auxiliary Systems</u>

3.3.2.2.1 <u>Loss of Material due to Galvanic, General, Pitting and Crevice Corrosion</u>

RAI 3.3.2.2.1-1
LRA Table 3.3-1 AMR Item 6 states that the ambient environment at VCSNS does not contain contaminants of sufficient concentration to cause any applicable aging effects requiring aging management for stainless steel components exposed to moist air environment. More information is needed to evaluate the applicant's determination that there are no aging effects for these stainless steel components. Submit information about the concentration of contaminants in the VCSNS ambient environment and present technical basis for determining an absence of aging effects requiring aging management.

4.2 <u>Reactor Vessel Neutron Embrittlement</u>

4.2.2.1 <u>Upper Shelf Energy</u>

RAI 4.2.2.1-1
Review of the reactor vessel integrity database (RVID) indicates that the applicant has submitted the following results, dated January 1999, for 32-EFPY USE for the VCSNS reactor vessel beltline materials. The applicant reports the 32-EFPY USE value of 74.1 ft-lb for

intermediate shell plate A9154-1 and 70.5 ft-lb for the lower shell plate C9923-2. It appears that plate A9154-1 may not be a limiting material as far as USE value is concerned. Submit technical basis for plate A9154-1 being a limiting material. Also explain why the 32-EFPY value for plate A9154-1 reported in January 1999 is higher than the one (67.5 ft-lb) reported in June 1992.

RAI 4.2.2.1-2
Submit a table of the VC Summer 60 year EOL USE values for each of the beltline materials. Tabulate the Initial USE, the EOL 1/4 T fluence, and the EOL 1/4 T USE.

4.2.2.2 Pressurized Thermal Shock

4.2.2.2-1 Submit a table of the VC Summer 60 year EOL RT_{PTS} values for each of the beltline materials. Tabulate the chemistry factor, Initial RT_{NDT} , margin, EOL peak fluence, fluence factor, delta RT_{PTS}, and EOL RT_{PTS}.

4.2.2.3 Pressure-Temperature Limits

No questions

Other TLAAs

4.7.1 Reactor Coolant Pump Flywheel

RAI 4.7.1-1
The staff-approved version of WCAP-14535, "Topical Report of Reactor Coolant Pump Flywheel Inspection Elimination," was published as WCAP-14535A in November 1996. This report also includes the staff's Requests for Additional Information (RAIs) and Safety Evaluation Report (SER) for WCAP-14535. The applicant states that WCAP-14535A allows the elimination of RCP flywheel inspections. However, the information presented in WCAP-15666,"Extension of Reactor Coolant Pump Motor Flywheel Examination," Rev. 0, Non-Proprietary Class 3, July 2001, contradicts the applicant's statement. According to WCAP-14666, the staff's SER for WCAP-14535 does not allow total elimination of inspections. In addition, the applicant states that WCAP-14535A supports the elimination of RCP flywheel inspections based on the insignificant increase in probability of failure achieved by inspections over a 60-year service life. However, according to WCAP-15666, the staff has stated in the SER for WCAP-14535 that they had not reviewed the risk assessment in WCAP-14535, but solely relied on the deterministic methodology to review the submittal. Clarify this discrepancy.

RAI 4.7.1-2
10 CFR 54.21(c)(i) and (ii) discuss analyses required as part of the time-limited aging analysis (TLAA). In order to confirm that the applicant has satisfied the regulatory requirements, the staff needs to review these analyses. Please provide the analyses and provide any references that indicate that they have been previously reviewed by the NRC.

4.7.2 Leak-Before-Break (LBB)

4.7.2-1 As a result of the V.C. Summer event in which primary water stress corrosion cracking (PWSCC) was identified in an Inconel 82/182 main coolant loop-to-reactor pressure vessel weld, the NRC staff has become concerned about the impact of PWSCC on licensee leak-before-break (LBB) evaluations. NUREG-1061, Volume 3, "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee, Evaluation of Potential for Pipe Breaks," which addresses the general methodology accepted by the NRC staff for demonstrating LBB behavior, stipulates that no active degradation mechanism may be present in a line which is under consideration for LBB. Draft Standard Review Plan 3.6.3, "Leak-Before-Break Evaluation Procedures," suggests that lines with potentially active degradation mechanisms may be considered for LBB approval provided that two mitigating action/programs are in place to address the potential active degradation mechanism.

The NRC considers the resolution of the impact of PWSCC on existing LBB evaluations to be a 10 CFR Part 50, operating reactor issue. The NRC staff has previously addressed this issue with te industry's PWR Materials Reliability Project (MRP) and received an interim report from the MRP, "PWR Materials Reliability Project, Interim Alloy 600 Safety Assessment for U.S. PWR Plants (MRP-44), Part 1: Alloy 82/182 Pipe Butt Welds," dated April 2001, which attempted to provide a technical basis for addressing this issue. The NRC expects to receive a final version of the MRP-44, Part 1 report from the MRP. Based on the information in the final MRP report and any additional, relevant information available to the NRC staff, the NRC will evaluate what actions or analyses, if any, may be required to confirm the continued applicability of existing licensee LBB evaluations.

(a) Regarding the VC Summer LRA, the NRC staff requests that the applicant provide a licensee commitment which states that for the period of extended operation of VC Summer, the applicant will implement actions or perform analyses, as deemed to be necessary by the NRC, to confirm continued applicability of existing VC Summer LBB evaluations. These actions or analyses will be consistent with those required to address the impact of PWSCC on existing LBB evaluations under 10 CFR Part 50 considerations.

(vii) Identify all welds in reactor coolant pressure boundary piping approved for LBB which contain Inconel 82/182 material that are exposed to the reactor coolant system environment.

(viii) Present information about any mitigative actions (e.g., mechanical stress improvement) that may have taken place at VCSNS since submittal of the LRA to manage PWSCC cracks in Alloy 82/182 piping welds? If so, confirm whether the future VCSNS LBB analysis will account for these mitigative actions.

Aging Management Evaluations in the GALL Report that Are Relied on for License Renewal, For Which GALL Recommends Further Evaluation

3.1.2.2.2 Crack Initiation and Growth due to SCC

See RAI 3.1.2.4.2-3

3.1.2.2.3 Crack Initiation and Growth due to Stress Corrosion Cracking, Intergranular Stress Corrosion Cracking, and Thermal and Mechanical Loading

RAI 3.1.2.2.3-1
In LRA Appendix B.2.7, the applicant has identified the information sources that will be used to identify the susceptible locations in small-bore RCS piping and to select the sample locations for inspections. Confirm whether it will follow all on-going industry activities related to failure mechanisms for small-bore piping including the recommendations of the EPRI sponsored Materials Reliability Project (MRP) Industry Task Group (ITG) on Thermal Fatigue and will evaluate changes to inspection activities based on industry recommendations. Confirm also whether the samples locations selected for inspection will be the bounding locations for Class 1 small-bore piping within the scope of the license renewal.

3.1.2.2.4 Crack Initiation and Growth due to Stress Corrosion Cracking or Primary Water Stress Corrosion Cracking

RAI 3.1.2.2.4-1
In LRA Table 3.1-1, AMR Item 9, the applicant credits the Alloy 600 aging management program (LRA Appendix B.1.1) as one of the three programs for managing crack initiation and growth due to PWSCC of two Ni alloy components, core support pads and bottom head penetrations. In LRA Appendix B.1.1, the applicant also states that the Alloy 600 aging management program is consistent with GALL AMP XI.M11. However, according to Table 3.1-1 of NUREG-1800, the GALL AMP XI.M11 is credited for managing cracking only in CRD nozzles (i.e., vessel head penetrations) and no other Ni alloy components. Clarify this discrepancy.

RAI 3.1.2.2.4-2
The inservice inspection plan (LRA Appendix B.1.7) specifies ASME Section XI VT-3 examination to detect cracking of the core support pads. However, VT-3 examinations may not be sufficient to detect cracking of the core support pads. Submit an aging management program for managing cracking in core support pads and bottom head penetrations during the extended period of operation. Specifically, submit following information: (1) inspection method used in detecting cracking in these components, (2) technical basis showing adequacy of this method to detect cracking, (3) inspection frequency and its justification, and (4) acceptance criteria.

RAI 3.1.2.2.4-3
In LRA Table 3.1-1, AMR Item 11, the applicant credits LRA Appendix B.1.1, Alloy 600 aging management program, which is a condition-monitoring program, for managing cracking of Alloy 82/182 welds for the pressurizer instrumentation penetrations and heater sleeves due to PWSCC. However, in LRA Appendix B.1.1, the applicant states that this program is consistent with NUREG/CR-1801 AMP XI.M11, which includes only reactor pressure vessel head penetrations in its scope and no other Alloy 600 components. Therefore, the applicant needs

to modify the scope of its program (LRA Appendix B.1.1) to include all other Alloy 600 components in addition to reactor vessel head penetrations.

RAI 3.1.2.2.4-4
The applicant has not presented aging management review results for pressurizer heater sheaths. Confirm whether the heater sheaths at VCSNS are made of Alloy 600. If so, then provide a program for managing cracking of the heater sheaths due to PWSCC.

3.1.2.2.5 Loss of Fracture Toughness due to Neutron Irradiation Embrittlement

RAI 3.1.2.2.5-1
The reactor vessel internals program assumes sufficient redundancy in bolt function that the plant can continue to function safely with a fewer than 100% of the bolts intact. The staff finds that this approach is consistent with the one described in NUREG-1801, GALL Section XI.M16 ("PWR Vessel Internals"). However, the staff needs additional information for evaluating the program. With respect to the application of this program to the detection of irradiation embrittlement of the baffle former bolts, identify the neutron fluence threshold for which the baffle former bolts become susceptible to loss of fracture toughness due to neutron irradiation embrittlement and void swelling, submit technical justification for the threshold, identify the percentage of the bolts to be selected for inspection, and submit the technical basis for this selection process.

3.1.2.2.6 Crack Growth due to Cyclic Loading

RAI 3.1.2.2.6-1
In LRA Table 3.1-1, AMR Item 7, the applicant states that the VCSNS vessel is constructed of ASME SA 533 Grade B, Cl 1 plate material and not ASME SA 508 Cl 2 forgings. Therefore, the aging effect of growth of underclad crack is not applicable to the VCSNS vessel. However, Table 5.2-8 of the VCSNS FSAR identifies SA 508 Cl 2 as one of the materials for reactor vessel shell, flange and nozzle forgings and nozzle safe ends. Clarify this discrepancy.

3.1.2.2.7 Crack Initiation and Growth due to SCC and Irradiation-Assisted Stress Corrosion Cracking (IASCC)

RAI 3.1.2.2.7-1
Under the reactor vessel internals inspection program, the applicant has selected volumetric inspection as the plant-specific basis for addressing the issue of crack initiation and growth due to SCC and irradiation-assisted stress corrosion cracking in the baffle former bolts. However, the staff needs additional information for evaluating the program. With respect to the application of this program to the detection of IASCC cracking of the baffle former bolts, identify the neutron fluence threshold for which the baffle former bolts become susceptible to IASCC cracking, submit technical justification for the threshold, identify the percentage of the bolts to be selected for inspection, and submit the technical basis for this selection process.

Aging Management Programs (System Specific)

Small-Bore Class 1 Piping Inspection

RAI 3.1.2.3.1-1 This RAI is the same as RAI 3.1.2.2.3-1.

Alloy 600 Aging Management Program

RAI 3.1.2.3.2-1
The Alloy 600 management program (LRA Section B.1.1) relies on detecting PWSCC cracks in head penetrations by means of inspection for signs of boric acid leakage during outages. Submit the following additional information regarding the boric acid leakage inspection: (1) Confirm that the boric acid leakage inspection includes inspection of bare vessel head (insulation being removed from the vessel head prior to inspection), (2) confirm that after the inspection vessel head is cleaned of any boric acid deposits prior to installing the insulation, (3) confirm whether ASME VT-2 examination method is used to detect leakage through a crack in the vessel head penetration, and (4) since the leakage through a PWSCC crack is generally very small, provide technical basis ensuring that the boric acid leakage inspection will be able to detect such a small leakage.

RAI 3.1.2.3.2-2
In LRA Section B.1.1, the applicant states that PWSCC cracks can also be detected by monitoring primary coolant leakage per Technical Specifications during plant operation. It is unlikely that monitoring of primary coolant leakage would be sensitive enough to detect a very small leakage through a PWSCC crack. Submit operating experience supporting the use of primary coolant leakage monitoring during operation for detecting a PWSCC crack.

RAI 3.1.2.3.2-3
In response to NRC Bulletin 2001-01, the applicant states that, in 1999, VCSNS performed VT-3 inspections of the vessel head interior surface and did not find any recordable indications. Identify the objective of that inspection and confirm whether it can reliably detect any cracking or loss of material at the vessel head interior surface.

RAI 3.1.2.3.2-4
Confirm whether any plant-specific augmented inspection program for vessel head penetration, including a combination of surface and volumetric inspections, was developed in response to NRC GL 97-01.

RAI 3.1.2.3.2-5
NRC Bulletin 2002-2 was issued after the submittal of the LRA. Describe how the discussion in this bulletin on adequacy of the current inspection requirements and programs for vessel head penetrations would impact the VCSNS Alloy 600 aging management program.

RAI 3.1.2.3.2-6
As suggested in the NRC closure letter from K. R. Cotton to G. J. Taylor, dated December 17, 1999 for SCE&G response to Generic Letter 97-01; the LRA needs to include a summary of the results of any inspections that have been completed on VCSNS vessel head penetrations prior to the license renewal application. Therefore, submit the following information for these inspections: (1) number of vessel head penetrations inspected and their locations on the vessel head, (2) inspection methods used, (3) number of Alloy 82/182 attachment welds inspected, and (4) inspection results.

RAI 3.1.2.3.2-7
The applicant states that the program will be enhanced according to the changes indicated by emerging regulatory requirements and identified by industry programs. However, the Alloy 600 aging management program, described in LRA Section B.1.0, does not specify whether the applicant would participate in the industry program for managing PWSCC type aging on vessel head penetrations. Confirm VCSNS participation in the industry program.

RAI 3.1.2.3.2-8
The FSAR supplement for the Alloy 600 management program (LRA Section B.1.1) is presented in LRA Appendix A, Section 18.2.4. The supplement states that the pressurizer and steam generator subcomponents in addition to the vessel subcomponents are within the scope of the program. However, because the applicant states that the program is consistent with the GALL AMP XI.M11, the scope of the program is limited to only vessel head penetrations and does not include other Alloy 600 components. Clarify this discrepancy and modify the supplement accordingly.

Reactor Vessel Surveillance Program

RAI 3.1.2.3.3-1
In accordance with ASTM E185, for current 40-year practice, it is recommended that the last capsule to be removed should receive the same or higher fluence than the peak EOL fluence. Therefore, the applicant should confirm that the updated capsule removal schedule reflects a capsule to be withdrawn with a predicted fluence equal to or greater than the peak EOL fluence for the extended period of operation for VC Summer.

RAI 3.1.2.3.3-2
Provide information regarding the fluence calculation methodology, i.e., is it consistent with the recommendations of DG-1053 and RG 1.190? In addition, confirm whether an alternative dosimetry will be used at VCSNS to monitor neutron fluence during the period of extended operation.

3.1.2.3.4 Reactor Head Closure Studs Program

RAI 3.1.2.3.4-1
In LRA Table 3.1-1, AMR Item 18, the applicant states that the aging effect requiring management is loss of closure integrity rather than cracking of closure studs. Describe the difference between these two aging effects and justify the selection of loss of closure integrity as an aging effect requiring management.

RAI 3.1.2.3.4-2
The applicant states that this program manages the effects of stress relaxation in the reactor head closure studs. Explain how this program manages the effects of stress relaxation.

3.1.2.3.5 Reactor Vessel Internals Inspection

RAI 3.1.2.3.5.2-1
In LRA Appendix B, Section B.2.4, the applicant describes its AMP to manage aging processes in reactor vessel internals. The LRA states that this AMP is consistent with GALL

AMP XI.M16, with the clarification that the resolution criterion for the enhanced VT-1 examination at the Summer Plant is expected to be less than 0.0005-in. resolution, which is specified in the GALL program. Submit technical justification for using less than 0.0005-in resolution, and explain how the anticipated reduction in the resolution criterion will be determined.

LRA Section B.2.4 describes the VCSNS Reactor Vessel Internals Inspection Program, a new program to assess the condition of reactor vessel internals in order to assure that the applicable aging effects will not result in loss of the intended functions during the period of extended operation. With respect to the irradiation embrittlement of the RV internal components, the staff notes that NUREG/CR-6048, Oak Ridge National Laboratory, has used $5x10^{20}$ neutrons/cm^2 (E > 0.1 MeV) as the threshold for loss of fracture toughness due to radiation embrittlement in Type 304 austenitic stainless steel materials. Confirm whether this threshold value will be used at VCSNS for austenitic stainless steel vessel internals. If an alternate value is proposed, then submit a technical basis for that alternate value. Also provide the technical basis for the selection of the RV internal components for inspection. This RAI is similar to RAI 3.1.2.2.5-1.

RAI 3.1.2.3.5-3

LRA Section B.2.4 describes the VCSNS Reactor Vessel Internals Inspection Program, a new program to assess the condition of reactor vessel internals in order to assure that the applicable aging effects will not result in loss of the intended functions during the period of extended operation. With respect to the application of this program to the detection of irradiation-assisted stress corrosion cracking of RV internals components, the staff requests additional information on how the applicant determines which RV internal components are susceptible to irradiation-assisted stress corrosion cracking, what components will be selected for inspection, and what the technical basis is for this selection process. This RAI is similar to RAI 3.1.2.2.7-1.

RAI 3.1.2.3.5-4

LRA Section B.2.4 describes the VCSNS Reactor Vessel Internals Inspection Program, a new program to assess the condition of reactor vessel internals in order to assure that the applicable aging effects will not result in loss of the intended functions during the period of extended operation. The staff reviewed the applicant's FSAR supplement (LRA Section 18.2.18) to verify that it provides an adequate description of the programs credited with managing this aging effect, as required by 10 CFR 54.21(d). The staff notes that the description of the Reactor Vessel Internals Inspection in LRA Section B.2.4 states that "specific acceptance criteria for changes in dimension due to void swelling, loss of preload due to stress relaxation, and loss of material due to wear will be determined by analysis as part of the inspection plan." The staff requests that the applicant commit to supplement the reactor vessel internals inspection program and to submit an integrated report to thte NRC prior to the end of the initial operating term for VC Summer. The report should summarize the understanding of the aging effects applicable to the reactor vessel internals and should contain a description of the VC Summer inspection plan, including methods for detection and sizing of cracks and acceptance criteria. This should also be discussed in the FSAR supplement

3.1.2.3.6 Bottom-Mounted Instrumentation Inspection

RAI 3.1.2.3.6-1
The applicant states that there is no NUREG-1801 (GALL report) item addressing the bottom-mounted instrumentation inspection program. This is not so. Item IV.B2.6-c of NUREG-1801 refers to the recommendations of NRC I&E Bulletin 88-09, "Thimble Tube Thinning in Westinghouse Reactors." Confirm whether the bottom-mounted instrumentation inspection program follows all the recommendations of the bulletin. If not, then submit a justification for not following these recommendations.

RAI 3.1.2.3.6-2
The applicant states that the bottom-mounted instrumentation inspection program monitors tube wall degradation in 100% of the BMI thimble tubes using eddy current testing. Submit information about whether the entire length of each thimble tube is inspected or only a selected portion of the length and present corresponding technical basis.

RAI 3.1.2.3.6-3
The applicant states that the frequency of ECT examination is based on an analysis of data obtained using wear rate relationships that are predicted based on Westinghouse research. Submit an explanation for the wear rate relationship and describe the Westinghouse research mentioned here.

RAI 3.1.2.3.6-4

(a) The applicant states that the ECT results are trended, wear rates are calculated, and inspections are planned prior to the refueling outage in which the thimble tube wear is predicted to exceed the acceptance criteria. Regarding the predicted wear rate, NRC I&E Bulletin 88-09 states that, based on the available data, it is not possible to accurately predict thimble tube wear rates. Explain how this difficulty in accurately predicting thimble tube wear rates is taken into account in developing the applicant's plan for the next thimble tube inspection.

(b) In describing its operating experience, the applicant states that the analysis of the wear rate data derived from the inspections performed at RF-4 and RF-5 determined that the next inspection of the thimble tubes is not required until RF-14. Explain and justify the use of this extrapolation of the limited inspection results for scheduling the next inspection of the thimble tubes.

RAI 3.1.2.3.6-5
The bottom-mounted instrumentation inspection program uses 75% loss of initial wall thickness as an acceptance criterion. Provide the technical justification for this criterion and explain how the allowances for such items as inspection methodology and wear scar geometry uncertainties, which were mentioned in NRC I&E Bulletin 88-09, are included in the criterion.

RAI 3.1.2.3.6-6

(a) The bottom-mounted instrumentation inspection program requires that the thimble tubes must be capped or repositioned if the projected through-wall wear exceeds 75% prior to the next scheduled ECT. Explain the factors that determine whether an affected thimble tube is to be capped or repositioned. If an affected tube is to be repositioned, then explain how the distance for repositioning is determined.

(b) The bottom-mounted instrumentation inspection program also requires that the thimble tube must be capped or replaced if projected through wall wear exceeds 80% prior to the next scheduled ECT. Explain the factors that determine whether an affected thimble tube is to be capped or replaced.

RAI 3.1.2.3.6-7
Since the issuance of IE Bulletin 88-09, the applicant has performed two inspections (RF-4 and RF-5) on thimble tubes at VCSNS. The applicant reports that several thimble tubes were repositioned in RF-5, but no thimble tubes were capped or required replacement. Confirm whether all the thimble tubes were inspected during these two inspections, and explain how it was determined to reposition several of these tubes.

Aging Management of Plant-Specific Components

3.1.2.4.1 Reactor Coolant System Non-Class 1 Components

RAI 3.1.2.4.1-1
LRA Table 3.1-2, AMR Item 14, states that stainless steel piping and piping system components including non-Class 1 pipe and valve bodies are internally exposed to treated water from the reactor makeup water system for pressurizer relief tank spray. As a result, these components are subject to loss of material due to crevice and pitting corrosion, and the chemistry program, LRA Appendix B.1.4, Chemistry Program is credited with managing this aging effect for these components. The applicant states that the chemistry program is similar to GALL AMP XI.M2, except it does not include inspection of selected components to verify the effectiveness of the program. The applicant further states that its evaluation of the applicable aging effect and the corresponding aging management program are consistent in material and environment, aging effect and credited program with the GALL Chapter VII.C2, “Closed Cycle Cooling Water System.” However, GALL AMP XI.21, “Closed Cycle Cooling Water System,” requires inspection to detect loss of material due to corrosion at locations of stagnant flow condition and crevices. Identify or submit an aging management program to verify the effectiveness of the chemistry program to manage loss of material.

RAI 3.1.2.4.1-2
According to LRA Table 3.3-1, AMR Item 14, the applicant credits the chemistry program (LRA Appendix B.1.4) for managing loss of material in RCP thermal barrier flange. The applicant states that the chemistry program is similar to GALL AMP XI.M21, except it does not include inspection. The use of the chemistry program alone may be inadequate because the program does not require inspection of the components to determine whether loss of material is taking place. The GALL AMP XI.M21 requires such inspection. Therefore, confirm that the effectiveness of the chemistry program is verified for the non-Class 1 RCP thermal barrier flange and piping/tubing

3.1.2.4.2 Reactor Coolant Piping, Valves and Pumps

RAI 3.1.2.4.2-1
LRA Table 3.1-1, AMR Item 22, identifies loss of closure integrity rather than loss of preload and cracking as an aging effect for stainless steel and low-alloy steel bolting requiring management. Explain how the management of loss of closure integrity instead of loss of preload and cracking would ensure that the intended function of the bolted joint (pressure boundary integrity) would be maintained during the extended period of operation.

RAI 3.1.2.4.2-2
LRA Table 3.1-1, AMR Item 20, states that the CASS elbows and nozzles of the RCS Class 1 piping are not susceptible to loss of fracture toughness because these components have low molybdenum content and have delta ferrite levels of less than 20%. This is acceptable because the material chemistry for these components meets the screening criteria set forth in the letter dated March 19, 2000, from Christopher Grimes, NRC, to Douglas Walters, NEI. This AMR Item is, however, not identified in LRA Table 2.3-2. Clarify this discrepancy.

RAI 3.1.2.4.2-3
The austenitic stainless steel RCS piping is susceptible to stress corrosion cracking at their external surface if it comes in contact with halogens that may be present in the thermal insulation. Cracking has not been identified as an aging effect at the external surface of these components. Confirm that all insulation used on austenitic stainless steel RCS piping to ensure that the piping is not susceptible to stress-corrosion cracking from halogens. Note that this is identified as License Renewal Action Item 4 by the industry report WCAP-14575-A, "Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components, December 2000."

RAI 3.1.2.4.2-4
LRA Table 3.1-1, AMR Item 22, identifies boric acid corrosion surveillances (LRA Appendix B.1.2) and the inservice inspection plan (LRA Appendix B.1.7) for managing aging effects in the RCS Class 1 bolted closures. Clarify how these two programs are sufficient to manage aging effects of loss of preload and cracking of RCS Class 1 bolting such that its intended function (pressure boundary) is maintained during the extended period of operation.

RAI 3.1.2.4.2-5
The NRC Information Notice 2000-17, "Crack in Weld Area of Reactor Coolant System Hot Leg Piping at V. C. Summer," reports a crack in Alloy 182/82 weld between the 'A' hot leg nozzle and stainless steel piping. Submit the following information related to this event:

(a) Explain how this cracking event has been taken into account in the ISI Plan (LRA Appendix B.1.7)

(b) The operating experience is described in LRA Appendix B.1.1, Alloy 600 Aging Management Program but it is not clear whether this program is credited for managing PWSCC cracking in Alloy 82/182 welds in RCS Class 1 piping. Clarify this.

(c) Identify any mitigative actions (e.g., mechanical stress improvement) taken since the submittal of the LRA to minimize the growth of existing PWSCC cracks and describe any plan for ensuring the effectiveness of these actions during the extended period of operation.

RAI 3.1.2.4.2-6
The chemistry program (LRA Appendix B.1.4) references water quality that is compatible with the materials of construction used in the Class 1 piping and associated components in order to minimize loss of material and cracking. This program incorporates EPRI and Institute of Nuclear Power Operations (INPO) guidelines, which reflect industry experience, and the "lessons learned" from VCSNS and external industry operating experience. Confirm whether the chemistry program incorporates the guidelines in EPRI TR-105714 (Rev. 3 or later revisions or update). Identify any differences between the chemistry program and these guidelines and submit technical justification for these differences.

RAI 3.1.2.4.2-7
According to LRA Table 2.3-2, the results for austenitic stainless steel piping and fittings (less than NPS 4"), and orifices exposed to chemically treated borated coolant are presented in LRA Table 3.1-1, AMR Item 6, and LRA Table 3.1-2, AMR Item 6. Both AMR items identify the same aging effect (i.e., cracking) but different aging management programs. AMR Item 6 of LRA Table 3.1-1 credits three programs, the chemistry program (LRA Appendix B.1.4), the ISI plan (LRA Appendix B.1.7), and the small-bore Class 1 piping inspections (LRA Appendix B.2.7) for managing cracking. However, AMR Item 6 of LRA Table 3.1-2 credits only one program, the chemistry program (LRA Appendix B.1.4), for managing cracking. Explain this apparent discrepancy.

3.1.2.4.3 Reactor Vessel

RAI 3.1.2.4.3-1
LRA Table 2.3-3 refers to LRA Table 3.1-1, AMR Item 24, for AMR results for stainless steel cladding on reactor vessel closure head dome, closure head and vessel flanges, and bottom head. The AMR item identifies cracking as an aging effect requiring management. However, the GALL report, NUREG 1801, does not identify cracking as an aging effect for cladding on these components, which are made of SA 533B, Cl 1. Clarify this inconsistency. In addition, submit technical basis for identifying cracking as an applicable aging effect for the cladding.

RAI 3.1.2.4.3-2
In LRA Table 3.1-1, AMR Item 22, the applicant states that loss of material due to wear is not considered a valid aging effect for control rod drive flange bolting requiring management. This statement implies that VCSNS has installed control rod drive flange bolting. However, Section 5.4.2 of the VCSNS FSAR states that the upper ends of the CRD nozzles have a welded flexible canopy seal and not bolting. Explain this discrepancy.

RAI 3.1.2.4.3-3
LRA Table 2.3-3 refers to LRA Table 3.1-1, AMR Item 28, for the AMR results for the reactor vessel closure studs assembly. However, LRA Table 3.1-1, AMR Item 28, presents the AMR results for vessel and vessel closure head flanges and not for closure studs assembly. Explain this discrepancy. (Note that according to the GALL report, the component group addressed by

the AMR Item 28 should have included reactor vessel and reactor vessel closure head flanges instead of reactor vessel closure studs.)

RAI 3.1.2.4.3-4
The austenitic stainless steel and Ni-alloy base reactor vessel appurtenances (i.e., CRD housings, vessel head penetrations, and Alloy 82/182 welds) are susceptible to stress corrosion cracking at the external surface if they come in contact with halogens that may be present in the thermal insulation. The applicant has not identified cracking as an aging effect at the external surface of these components. Submit a description of all insulation used on austenitic stainless steel and Ni-alloy base reactor vessel components and demonstrate that these components are not susceptible to stress-corrosion cracking from halogens.

RAI 3.1.2.4.3-7
The chemistry program (LRA Section B.1.4) also manages loss of material in stainless steel and Ni-alloy reactor vessel components (i.e., CRD housings, cladding, vent plug, bottom head and closure head penetration tubes, reactor vessel core support pads, and nozzle safe ends) internally exposed to chemically treated borated coolant. The staff does not believe that the chemistry program alone can effectively manage loss of material, especially due to crevice corrosion, in these components. The applicant needs to provide an AMP to ensure that no unacceptable loss of material is occurring in these components.

3.1.2.4.4 Reactor Vessel Internals

RAI 3.1.2.4.4-1
In LRA Table 3.1-1, AMR Items 5 and 31, the applicant identifies loss of fracture toughness due to irradiation as one of the applicable aging effects for the stainless steel reactor vessel internals in the fuel zone region. The applicant's identification of all of the reactor vessel internals in the fuel zone region as being susceptible to loss of fracture toughness due to irradiation represents an acceptable position to the staff. However, the staff needs additional information. Submit a criterion used to identify the vessel internals that are susceptible to loss of fracture toughness due to neutron irradiation along with its technical basis, and explain why the reactor vessel internals outside the fuel zone region are not considered susceptible to loss of fracture toughness due to irradiation.

RAI 3.1.2.4.4-2
The applicant credits the Reactor Vessels Internals Inspection Program (LRA Appendix B.2.4) alone with managing loss of preload due to stress relaxation in VCSNS hold-down spring, clevis insert bolts, and upper and lower support column bolts (LRA Table 3.1-1, AMR Items 30 and 35). In contrast, NUREG-1801 specifies that both inservice inspection and loose parts monitoring for managing loss of preload due to stress relaxation in the lower and upper support column bolts (GALL Items IV, B2.1-k and B2.5-h). For the hold-down spring (GALL Item B2.1-d) and clevis insert bolts (GALL Item IV, B2.5-i), NUREG-1801 states that either loose parts monitoring or neutron noise monitoring is to be used in addition to inservice inspection to manage loss of preload. Explain how the Reactor Vessels Internals Inspection Program alone in the absence of either loose parts monitoring or neutron noise monitoring will adequately manage loss of preload in these components.

3.1.2.4.5 In-core Instrumentation System

3.1.2.4.5-1 In LRA Table 3.1-2, AMR Item 4 identifies stainless steel as material for in-core thermocouple seal bolting. However, in Discussion column for this AMR item, the applicant refers to high strength material for this bolting. Clarify this discrepancy. If high-strength, low-alloy steel is the bolting material, then explain why loss of material due to boric acid corrosion caused by leaking borated coolant is not an aging effect for this bolting material.

3.1.2.4.5-2 The applicant has identified loss of closure integrity rather than loss of preload and cracking as an applicable aging effect requiring management for closure bolting for in-core thermocouple seals. Explain how managing of loss of closure integrity would ensure that the pressure boundary of the bolted joint would be maintained during the extended period of operation.

3.1.2.4.5-3 In Westinghouse-designed PWRs, mechanical high-pressure seals, located at the seal table, are used to seal the area between the thimble tubes and the long-radius guides. Describe how the sealing of the area between the thimble tube and the guide is achieved at VCSNS and confirm whether bolted connection is employed for this mechanical seal. If a bolted connection is employed, then identify applicable aging effects and present an AMP for managing these effects.

3.1.2.4.5-4 The inservice inspection plan program (LRA Section B.1.7) is credited for managing loss of mechanical closure integrity, which includes loss of preload, loss of material and cracking of the bolted closures for the in-core thermocouple seal assemblies. The inservice inspection plan, which is based on ASME Section XI, Subsection IWB, requires VT-1 visual examination of bolts. Explain how the VT-1 examination can manage the effect of loss of preload so that the intended function of the bolted closure, i.e., pressure boundary, is maintained during the extended period of operation.

3.1.2.4.6 Pressurizer

RAI 3.1.2.4.6-1
The applicant states that the identification of the applicable aging effects for the pressurizer in LRA Table 3.1-1 is consistent with the GALL report. However, The GALL report presents an AMR for five additional pressurizer components [pressurizer seismic lugs, heater elements (heater sheaths), manway pad gasket seating surface, safety valves, and relief valves] that are not addressed in the LRA. According to Table 2-1 in the Westinghouse report WCAP 14574-A “License Renewal Evaluation: Aging Management Evaluation for Pressurizers,” the first three components are within the scope of license renewal and require AMRs. Provide technical justification for not presenting an AMR for these three components, i.e., pressurizer seismic lugs, heater elements, and manway pad gasket seating surface. Also explain why the AMR results for safety and relief valves are not presented in LRA Table 3.1-1.

RAI 3.1.2.4.6-2
According to LRA Table 2.3-6, the applicant presented an AMR for pressurizer nozzles and safe ends. However, it is not clear to the staff about which specific nozzles are addressed by the LRA. Confirm whether the following five pressurizer nozzles and safe ends are included:

surge nozzle, spray nozzle, safety nozzle, relief nozzle, and their safe ends, and instrument nozzle.

RAI 3.1.2.4.6-3
In LRA Table 2.3-6, the applicant presented AMR of manway cover (Row 4) and manway forgings (Row 7) exposed to chemically treated borated coolant. Why are the AMR results for these two components different? Does the AMR of manway forgings include that of manway flanges?

RAI 3.1.2.4.6-4
According to Section 3.2.5 of the Westinghouse report, WCAP-14574-A, "License Renewal Evaluation: Aging Management Evaluation for Pressurizers," four components of the pressurizer for which an AMR is performed, are exposed to fluid flows that have the potential to result in erosion of the components: surge nozzle thermal sleeve and safe end, and spray nozzle thermal sleeve and safe end. The applicant has not identified loss of material due to erosion as an applicable aging effect for these components. Explain why loss of material due to erosion is not an applicable aging effect for these components. If loss of material due to erosion is an applicable aging effect, then provide an AMP for managing it.

RAI 3.1.2.4.6-5
The attachment welds at the inside surface of the pressurizer are susceptible to cracking due to stress corrosion cracking if they are sensitized during fabrication. The applicant has not presented an AMR for these welds. Identify the components that are welded to the inside surface of the pressurizer and provide technical justification for determining whether cracking due to SCC is an applicable aging effect. If cracking is an applicable aging effect for the attachment welds, then provide an AMP for managing this effect.

RAI 3.1.2.4.6-6
In LRA Table 3.1-2, AMR Item 7, the applicant credits the chemistry program (LRA Appendix B.1.4) for managing loss of material due to crevice and pitting corrosion in the pressurizer shell and heads cladding with austenitic stainless steel and stainless steel components internally exposed to chemically treated borated coolant. These components are susceptible to crevice and pitting corrosion because high levels of oxygen may be present in the PWR reactor coolant. However, PWR licensees maintain hydrogen overpressure in the reactor coolant, and if the overpressure were at sufficiently high levels, it would provide protection in creviced geometries on the internal surfaces of the pressurizer. Explain how the chemistry program will provide for a sufficient level of hydrogen overpressure to manage crevice corrosion of the internal surfaces of the pressurizer.

RAI 3.1.2.4.6-7
LRA Table 3.1-1, AMR Item 24, credits the chemistry program (LRA Appendix B.1.4) and in-service inspection plan program (LRA Appendix B.1.7) for managing cracking of the pressurizer shell, lower head and upper head cladding with austenitic stainless steel and internally exposed to chemically treated borated coolant. The in-service inspection plan is mainly directed at structural welds in the pressurizer shell and heads and not at stainless steel cladding. However, in 1990, crack-like indications were discovered in the Haddam Neck pressurizer cladding. Thermal fatigue can initiate and propagate such cracking through the cladding and into the ferritic base metal or weld metal beneath the clad. Therefore, submit an

AMP to verify whether thermal fatigue-induced cracking has initiated in the clad and propagated through it into the ferritic base metal or weld metal beneath the clad.

RAI 3.1.2.4.6-8
LRA Table 3.1-1, AMR Item 26, credits the boric acid corrosion surveillance program (LRA Appendix B.1.2) for managing loss of material due to boric acid corrosion of pressurizer carbon steel and low alloy steel components: shell, upper and lower heads, nozzles, integral support, and manway cover and bolts. Provide detailed information showing how the program will be sufficient to manage the corrosive effects of boric acid leakage on these components during the extended period of operation, including postulated leakage from the pressurizer nozzle-to-vessel welds, pressurizer nozzle-to-safe end welds, and pressurizer manway bolting materials.

RAI 3.1.2.4.6-9
LRA Table 3.1-1, AMR Item 22, credits the in-service inspection plan program (LRA Appendix B.1.7) for managing loss of mechanical closure integrity, which includes loss of preload, loss of material and cracking, of the bolted closures for the pressurizer manway cover bolts. The inservice inspection plan, which is based on ASME Section XI, Subsection IWB, requires volumetric, and VT-1 and VT-2 visual examinations of bolts. Explain how these examinations manage the effects of loss of preload so that the intended function of the bolted closure, i.e., pressure boundary, is maintained during the extended operation.

3.3 Auxiliary Systems

3.3.2.2.1 Loss of Material due to Galvanic, General, Pitting and Crevice Corrosion

RAI 3.3.2.2.1-1
LRA Table 3.3-1 AMR Item 6 states that the ambient environment at VCSNS does not contain contaminants of sufficient concentration to cause any applicable aging effects requiring aging management for stainless steel components exposed to moist air environment. More information is needed to evaluate the applicant's determination that there are no aging effects for these stainless steel components. Submit information about the concentration of contaminants in the VCSNS ambient environment and present technical basis for determining an absence of aging effects requiring aging management.

4.2 Reactor Vessel Neutron Embrittlement

4.2.2.1 Upper Shelf Energy

RAI 4.2.2.1-1
Review of the reactor vessel integrity database (RVID) indicates that the applicant has submitted the following results, dated January 1999, for 32-EFPY USE for the VCSNS reactor vessel beltline materials. The applicant reports the 32-EFPY USE value of 74.1 ft-lb for intermediate shell plate A9154-1 and 70.5 ft-lb for the lower shell plate C9923-2. It appears that plate A9154-1 may not be a limiting material as far as USE value is concerned. Submit technical basis for plate A9154-1 being a limiting material. Also explain why the 32-EFPY

value for plate A9154-1 reported in January 1999 is higher than the one (67.5 ft-lb) reported in June 1992.

RAI 4.2.2.1-2
Submit a table of the VC Summer 60 year EOL USE values for each of the beltline materials. Tabulate the Initial USE, the EOL 1/4 T fluence, and the EOL 1/4 T USE.

4.2.2.2 Pressurized Thermal Shock

4.2.2.2-1 Submit a table of the VC Summer 60 year EOL RT_{PTS} values for each of the beltline materials. Tabulate the chemistry factor, Initial RT_{NDT} , margin, EOL peak fluence, fluence factor, delta RT_{PTS}, and EOL RT_{PTS}.

4.2.2.3 Pressure-Temperature Limits

No questions

Other TLAAs

4.7.1 Reactor Coolant Pump Flywheel

RAI 4.7.1-1
The staff-approved version of WCAP-14535, "Topical Report of Reactor Coolant Pump Flywheel Inspection Elimination," was published as WCAP-14535A in November 1996. This report also includes the staff's Requests for Additional Information (RAIs) and Safety Evaluation Report (SER) for WCAP-14535. The applicant states that WCAP-14535A allows the elimination of RCP flywheel inspections. However, the information presented in WCAP-15666,"Extension of Reactor Coolant Pump Motor Flywheel Examination," Rev. 0, Non-Proprietary Class 3, July 2001, contradicts the applicant's statement. According to WCAP-14666, the staff's SER for WCAP-14535 does not allow total elimination of inspections. In addition, the applicant states that WCAP-14535A supports the elimination of RCP flywheel inspections based on the insignificant increase in probability of failure achieved by inspections over a 60-year service life. However, according to WCAP-15666, the staff has stated in the SER for WCAP-14535 that they had not reviewed the risk assessment in WCAP-14535, but solely relied on the deterministic methodology to review the submittal. Clarify this discrepancy.

RAI 4.7.1-2
10 CFR 54.21(c)(i) and (ii) discuss analyses required as part of the time-limited aging analysis (TLAA). In order to confirm that the applicant has satisfied the regulatory requirements, the staff needs to review these analyses. Please provide the analyses and provide any references that indicate that they have been previously reviewed by the NRC.

4.7.2 Leak-Before-Break (LBB)

4.7.2-1 As a result of the V.C. Summer event in which primary water stress corrosion cracking (PWSCC) was identified in an Inconel 82/182 main coolant loop-to-reactor pressure vessel weld, the NRC staff has become concerned about the impact of PWSCC on licensee leak-before-break (LBB) evaluations. NUREG-1061, Volume 3, "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee, Evaluation of Potential for Pipe Breaks," which addresses the general methodology accepted by the NRC staff for demonstrating LBB behavior, stipulates that no active degradation mechanism may be present in a line which is under consideration for LBB. Draft Standard Review Plan 3.6.3, "Leak-Before-Break Evaluation Procedures," suggests that lines with potentially active degradation mechanisms may be considered for LBB approval provided that two mitigating action/programs are in place to address the potential active degradation mechanism.

> The NRC considers the resolution of the impact of PWSCC on existing LBB evaluations to be a 10 CFR Part 50, operating reactor issue. The NRC staff has previously addressed this issue with te industry's PWR Materials Reliability Project (MRP) and received an interim report from the MRP, "PWR Materials Reliability Project, Interim Alloy 600 Safety Assessment for U.S. PWR Plants (MRP-44), Part 1: Alloy 82/182 Pipe Butt Welds," dated April 2001, which attempted to provide a technical basis for addressing this issue. The NRC expects to receive a final version of the MRP-44, Part 1 report from the MRP. Based on the information in the final MRP report and any additional, relevant information available to the NRC staff, the NRC will evaluate what actions or analyses, if any, may be required to confirm the continued applicability of existing licensee LBB evaluations.

(a) Regarding the VC Summer LRA, the NRC staff requests that the applicant provide a licensee commitment which states that for the period of extended operation of VC Summer, the applicant will implement actions or perform analyses, as deemed to be necessary by the NRC, to confirm continued applicability of existing VC Summer LBB evaluations. These actions or analyses will be consistent with those required to address the impact of PWSCC on existing LBB evaluations under 10 CFR Part 50 considerations.

(vii) Identify all welds in reactor coolant pressure boundary piping approved for LBB which contain Inconel 82/182 material that are exposed to the reactor coolant system environment.

(viii) Present information about any mitigative actions (e.g., mechanical stress improvement) that may have taken place at VCSNS since submittal of the LRA to manage PWSCC cracks in Alloy 82/182 piping welds? If so, confirm whether the future VCSNS LBB analysis will account for these mitigative actions.

VCSNS AUXILIARY SYSTEMS DRAFT RAIS

GENERAL RAIS

The following RAIs are applicable to several components in auxiliary systems and are, therefore, considered general RAIs for auxiliary systems.

RAI 3.3-1
Numerous tables included in the application list the component material and environment to which the component is exposed. However, the applicant did not provide a description of these environments in the LRA. It should be noted that the aging effect depends on the component material as well as the plant specific environment characteristic. A description of the specific information (such as ranges of temperature, humidity, and/or compositions etc.) related to the plant specific environment characteristic considered in the VCSNS LRA will provide the necessary environment information for the staff to perform its AMR of the components of the Auxiliary Systems well as other systems in the VCSNS. The applicant is requested to provide a description of these environments included in the LRA.

RAI 3.3-2
This common RAI concerns aging mechanisms related to the aging effect of loss of materials in sheltered environment for carbon steel components in auxiliary systems described below.

Gaseous Waste Processing System

In the table entitled "Virgil C. Summer Nuclear Station Database AMR Query", the applicant stated that for carbon steel in a sheltered environment, the aging effect of loss of material is due only to general corrosion. While in the AMR Query Note A-WG-C, the applicant stated that microbiologically induced corrosion (MIC) is also an applicable aging effect for carbon steel in a sheltered environment. The applicant is requested to clarify this discrepancy. In addition, it should be noted that the GALL report identifies the additional aging mechanisms of pitting and crevice corrosion in moist air and the additional aging mechanism of MIC in warm, moist air. Provide the justification on why these aging mechanisms have not been addressed.

Instrument Air Supply System

In the Database Query Table of VCSNS LRA, no aging effect is identified for the carbon steel components exposed to external environment of sheltered. For carbon steel components exposed to external environments of moist air such as sheltered environment, the GALL report identified that loss of material due to general, pitting, crevice corrosion and MIC is an aging effect. The applicant is requested to justify why loss of material due to general, pitting, crevice corrosion or MIC is not an applicable aging effect for the carbon steel components exposed to an external sheltered environment. Provide the technical basis for this conclusion.

RAI 3.3-3
This common RAI concerns the susceptibility to aging effects for stainless steel components in ambient environment in auxiliary systems described below.

Liquid Waste Processing System

Stainless steel components in ambient environment may be subject to loss of material aging effects due to pitting, crevice corrosion, and MIC. In the VCSNS AMR Database AMR Query Table, the applicant identified no aging effects for stainless steel piping/fitting and valve (body only) in reactor building and sheltered environments because of the presence of insignificant concentration of contaminants in these environments. Please provide the technical basis including the acceptance criterion and the verification/inspection activities on susceptible locations to justify this basis.

Nuclear & Non-nuclear Plant Drains

Stainless steel components in ambient environment may be subject to loss of material aging effects due to pitting, crevice corrosion and MIC. In the VCSNS Database AMR Query table, the applicant identified no aging effects for stainless steel piping/fittings and valves (body only) in reactor building or sheltered environments because of the presence of insignificant concentration of contaminants in these environments. Please provide the technical basis including the acceptance criterion and the verification/inspection activities on susceptible locations to justify this basis.

Roof Drains System

Stainless steel components in ambient environment may be subject to aging effects of loss of material due to pitting, crevice corrosion and MIC. In the VCSNS Database AMR Query table, the applicant identified no aging effects for stainless steel pipe/fittings in the reactor building environment because of the presence of insignificant concentration of contaminants in this environment. Please provide the technical basis including the acceptance criterion and the verification/inspection activities on susceptible locations to justify this basis.

Station Service Air System

Stainless steel components in ambient environment may be subject to aging effect of loss of material due to pitting, crevice corrosion and MIC. In the VCSNS Database AMR Query table, the applicant identified no aging effects for stainless steel pipe and fittings, tube and tube fittings, and valves (body only) in reactor building and sheltered environments because of the presence of insignificant concentration of contaminants in these environments. Please provide the technical basis including the acceptance criterion and the verification/inspection activities on susceptible locations to justify this basis.

<u>SYSTEM-SPECIFIC RAIS</u>

The followings are RAIs that are system-specific.

3.3.2.4.1 Air Handling and Local Ventilation and Cooling System

RAI 3.3.2.4.1-1
The table entitled "Virgil C. Summer Nuclear Station Database AMR Query" indicates that galvanized steel ductwork in a 'yard' environment has no identified aging effects and does not require an aging management program. The staff finds that this conclusion may not be justified because of factors associated with corrosive agents in the local environment and rainfall. Provide the justification for the conclusion that galvanized steel ductwork in a 'yard' environment has no identified aging effects.

RAI 3.3.2.4.1-2
The table entitled "Virgil C. Summer Nuclear Station Database AMR Query" states that carbon steel cooling coil headers in a treated water environment are subject to cracking due to stress corrosion cracking (SCC). However, no aging management program has been provided to address this aging effect. Explain why no aging management program has been provided to address this aging effect.

3.3.2.4.3 Building Services System

RAI 3.3.2.4.3-1
In the LRA Table 3.3-2, Item 11, the applicant stated that no aging effect was identified for the stainless steel piping and fittings in the air-gas environment. However, in the AMR Query Notes "A-BS-c" the applicant stated "Loss of material due to corrosive impacts of alternate wetting and drying are aging effects for stainless steel exposed to a ventilation environment, and subject to alternate wetting and drying that may concentrate contaminates. A review of the Air-Gas System Screening Report [TR00160-006], Attachment I and associated references determined that there are stainless steel components within the license renewal evaluation boundaries of the BS System which are exposed to alternative wetting and drying in the ventilation environment. Therefore, loss of materials and cracking due to corrosive impacts of alternative wetting and drying are not aging effects requiring management of stainless steel components/component types of the BS System exposed to the ventilation environment." Clarify with justifications the above quoted statements in LRA Table 3.3-2, Item 11 and the AMR Query Notes "A-BS-c".

3.3.2.4.4 Chilled Water System

RAI 3.3.2.4.4-1
LRA Table 3.3-2, Item 19 credits the Above Ground Tank Inspection program (B.2.1), and the Chemistry Program (B.1.4), for managing loss of material and cracking of the internal surfaces of the chilled water expansion tanks (XTK0174A/B) during the period of extended operation. The staff finds that this conclusion does not appear adequate to detect significant tank degradation in inaccessible locations such as tank bottom surfaces. Provide assurance that significant tank degradation in the inaccessible locations of these tanks such as tank bottom surfaces is adequately managed.

RAI 3.3.2.4.4-2
LRA Table 3.3-1, Item 5 credits the Inspections of Mechanical Components program (B.2.11) for managing loss of material of the external surfaces of the carbon steel chilled water

expansion tanks (XTK0174A/B) during the period of extended operation. The staff finds that this conclusion does not appear adequate to detect significant tank degradation in inaccessible locations such as under insulation or external tank bottom surfaces. Provide assurance that significant tank degradation in the inaccessible locations of these tanks such as under insulation or external tank bottom surfaces is adequately managed.

RAI 3.3.2.4.4-3
The table entitled “Virgil C. Summer Nuclear Station Database AMR Query” states that carbon steel components such as pump casings, evaporator tubesheets and water boxes, valve bodies, pipe and fittings, and tanks in a treated water environment are subject to cracking due to stress corrosion cracking (SCC). However, no aging management program has been provided to address this aging effect. Justify the lack of an aging management program.

3.3.2.4.6 Component Cooling Water System

RAI 3.3.2.4.6-1
Selective leaching is known to affect copper-nickel in aqueous environments with nickel being the element removed. Preventive measure involves proper selection of alloy/environment combination. For Copper-nickel components in treated water environment the applicant stated in the table entitled ”Virgil C. Summer Nuclear Station Database AMR Query” that loss of material due to selective leaching was determined not to be an aging effect for VCSNS. Provide the basis for this conclusion, including specific information on materials composition and environmental conditions that enable the applicant to draw this conclusion.

RAI 3.3.2.4.6-2
For stainless steel component in reactor building environment the applicant stated that for VCSNS no aging effects were determined to require aging management during the period of extended operation. Provide the basis of this conclusion. In particular, in view of the operational experience described in IN 85-30: Microbiologically Induced Corrosion of Containment Service Water System, explain why MIC is not an applicable aging mechanism leading to loss of material as applicable aging effect in VCSNS reactor building environment. In addition, for stainless steel component in sheltered environment the applicant stated that for VCSNS no aging effects of loss of material due to pitting and crevice corrosion were determined to require aging management. Provide the basis for this conclusion.

RAI 3.3.2.4.6-3
The applicant identified galvanic corrosion as an applicable aging effect for carbon steel component in treated water environment and the Chemistry program is stated to be the applicable AMP. It should be noted that the likely material/locations determining galvanic corrosion rates depends on which specific metal/alloy, how far apart the two dissimilar metals are on the galvanic series chart, the electrolyte conductivity, geometric factors and immersion time. Given these factors provide the basis that the Chemistry program is the applicable AMP for galvanic corrosion.

RAI 3.3.2.4.6-4
he applicant credited its Chemistry Program (which explicitly exempts the one-time inspection) for managing loss of material and cracking aging effects for some sub-components in heat exchangers in several auxiliary systems (e.g., tubes in a heat exchanger in CCWS, Page 31

of 413 of Database AMR Query). The applicant is requested to explain how the credited Chemistry Program alone will ensure the heat transfer function of the sub-components in the heat exchanger.

3.3.2.4.7 Diesel Generator Services Systems

RAI 3.3.2.4.7-1
LRA Table 3.3-1, Item 2 states that loss of material due to wear is not considered an aging effect because mechanical components must perform their License Renewal intended functions without moving part. Wear that occurs on non-moving components is considered to be caused by improper design and should be corrected by normal maintenance activities. The staff disagrees with the applicant's explanation that wear is caused by improper design in the non-moving components. The staff believes that wear of elastomer may be attributed to many conditions such as relative movement due to thermal expansion. The applicant is requested to provide the technical basis to justify why the aging effect of loss of material due to wear is not applicable.

RAI 3.3.2.4.7-2
No aging management program has been provided by the applicant for managing loss of material due to galvanic corrosion for any applicable components in the Diesel Generator Service Systems. Provide the basis for not including such an aging management program.

3.3.2.4.12 Instrument Air Supply System

RAI 3.3.2.4.12-1
For the aging management review of several components within the license renewal evaluation boundary of the instrument air supply system, the applicant stated that they are exposed to an oil-free, filtered, and dried compressed air (referred to as an air-gas environment) and loss of material is not an aging effect requiring management during the period of extended operation. It should be noted that in the instrument air system, components that are located upstream of the air dryers are generally exposed to a wet air/gas environment and, therefore, may be subject to loss of material due to general and pitting corrosion. In addition, it is reasonable to assume that components downstream of the dryers are exposed to dry air/gas environment. However, this may not be supported by some operating experience. For example, NRC IN 87-28, "Air Systems Problems at U.S. Light Water Reactors," provides the following: "A loss of decay heat removal and significant primary system heat up at Palisades in 1978 and 1981 were caused by water in the air system." This experience implies that the air/gas system downstream of the dryer may not be dry. On the basis of this industry experience, the applicant is requested to discuss its plant-specific operating experience related to components that are exposed to an instrument air environment, and to provide a technical basis for not identifying loss of material as an aging effect for these components.

3.3.2.4.14 Liquid Waste Processing System

RAI 3.3.2.4.14-1
The GALL report identifies the aging effects for carbon steel and stainless steel components exposed to treated water and corresponding aging management programs and recommends further evaluations. In the table entitled "Virgil C. Summer Nuclear Station Database Query",

the applicant asserts that the aging effects for the combination of these components/component types and environments are consistent with GALL and GALL does not recommend further evaluation. Explain the differences.

RAI 3.3.2.4.14-2
The GALL report identifies cracking due to stress corrosion cracking aging effects for stainless steel components exposed to treated water and corresponding aging management programs and recommends further evaluations. In the table entitled ”Virgil C. Summer Nuclear Station Database Query”, the applicant states that the aging effects for the combination of those components/component types and environments are consistent with GALL and GALL does not recommend further evaluations. Explain why the conclusion in the LRA is different from the GALL.

RAI 3.3.2.4.14-3
In the AMR query notes item A-WL-k, the applicant stated, “Some component surfaces such as the area around cooling coils are subject to alternate wetting and drying and are thus susceptible to pitting and crevice corrosion and stress corrosion cracking. This mechanism is not expected to be a significant degree in the ventilation air environment. The subject valve is not in a wetted location for the majority of the time and is considered to be dry during normal operation. As such, loss of material/cracking due to corrosive impacts of alternate wetting and drying are not aging effects requiring management.” Is there any inspection result that can support this conclusion? If not, provide justifications for this conclusion.

3.3.2.4.16 Nuclear Sampling System

RAI 3.3.2.4.16-1
For carbon steel components exposed to external environments of moist air such as reactor building or sheltered, the GALL report identified that loss of material is an aging effect that is caused by general, pitting, crevice corrosion and MIC. The VCSNS LRA identifies loss of material as an aging effect due to general corrosion only. Justify why pitting, crevice corrosion or MIC does not occur for the carbon steel components exposed to external environments of moist air such as reactor building or sheltered. If insignificant concentration of contaminants is part of the technical basis provide the acceptance criterion and the verification/inspection activities justifying this basis.

RAI 3.3.2.4.16-2
The nuclear sampling system contains borated water. However, the VCSNS B.1.2 Boric Acid Corrosion Surveillance AMP is not mentioned in the database AMR Query table of nuclear sampling system. Address how the loss of materials from boric acid corrosion due to borated water leakage is managed for the components of nuclear sampling system or provide the basis for why this is not an applicable aging effect.

3.3.2.4.17 Radiation Monitoring System

RAI 3.3.2.4.17-1
The table entitled “Virgil C. Summer Nuclear Station Database AMR Query” states that for stainless steel pipe and fittings in a sheltered environment, the loss of material due to MIC can be managed for the period of extended operation by the applicant’s Maintenance Rule

Structures Program (B.1.18). The applicant also stated that exposure of other stainless steel components; such as pressure retaining instrumentation, tanks, tube and tube fittings and valve bodies to the same sheltered environment has no aging effect. Address and clarify this inconsistency.

3.3.2.4.18 Reactor Makeup Water Supply System

RAI 3.3.2.4.18-1
In the table entitled "Virgil C. Summer Nuclear Station Database AMR Query", the applicant stated that for stainless steel pipe and fittings in a sheltered environment, the loss of material due to MIC can be managed for the period of extended operation by the applicant's Maintenance Rule Structures Program (B.1.18). The applicant also stated that exposure of other stainless steel components; such as orifices, pump casings, tube and tube fittings and valve bodies to the same sheltered environment has no aging effect. The applicant is requested to clarify this inconsistency.

3.3.2.4.20 Station Service Air System

RAI 3.3.2.4.20-1
Normally station service air system may contain elastomer materials in hose connection seals, duct seals, flexible collars between ducts and fans, rubber boots, etc. For some plant designs, elastomer components are used as vibration isolators to prevent transmission of vibration and dynamic loading to the rest of the system. The aging effects on those elastomer components are hardening and loss of material. However, no elastomer component associated with the station service air system was listed in the LRA. Clarify whether there are elastomer components present in the Station Service Air System and if so, address the management of the aging effects of hardening and loss of material on the elastomer components.

RAI 3.3.2.4.20-2
Loss of material due to boric acid corrosion for components adjacent to a source of borated water is an aging effect for carbon steel components. In the VCSNS Database AMR Query table, the applicant identified some carbon steel components in the reactor building and sheltered environments are subject to such an aging effect and some are not. Explain why different conclusions are attained for components with the same material/environment combination.

3.3.2.4.21 Service Water System

RAI 3.3.2.4.21-1
The applicant stated in the VCSNS Database AMR Query Table that galvanic corrosion is one of the applicable aging mechanism that give rise to the aging effect of loss of materials. The component group affected in this category for the Service Water System includes carbon steel couplings, and pipe and fittings in an underground environment. The Buried Piping and Tanks Inspection is stated as the applicable AMP. The applicant further stated that this AMP will be consistent with XI.M34, Buried Piping and Tanks Inspection, as identified in NUREG -1801 prior to the period of extended operation. The likelihood and extent of galvanic corrosion depends on the relative position of the contacting metal/alloys on the galvanic potential chart, the electrolyte, immersion time and geometrical factors. Many of these factors are location-

dependent. In B.2.10, Buried Piping and Tanks Inspection, in the applicant's LRA are inspections to be performed in areas with the highest likelihood on galvanic corrosion, and in areas with a history of corrosion problems? Or are inspections to be performed on an opportunistic basis? Provide justifications for either case.

RAI 3.3.2.4.21-2

For carbon steel component in the sheltered and reactor building environments of VCSNS is loss of materials from aging mechanisms other than boric acid corrosion (such as general corrosion, galvanic corrosion) an applicable aging effect? If so, identify the applicable aging effects and the associated AMPs. Or provide the technical basis to justify no other applicable aging effects for these components.

RAI 3.3.2.4.21-3

The Query Notes (A-SW-f) states that "Loss of material due to MIC is an aging effect for stainless steel components, and is a potential problem in sheltered environments where contamination from untreated water or soil may have introduced bacteria. VCSNS operating experience has identified the accumulation of microbiological organisms on the external surfaces of some piping components at building wall penetrations as a result of groundwater intrusion effects. The VCSNS AMR has conservatively considered all piping, process tubing and ductwork component types to be susceptible to external MIC if they either enter a building from the outside or pass between buildings included in the sheltered environment below the 425' elevation. Loss of material due to MIC is only an aging effect requiring management for the stainless steel process tubing which passes between buildings below the 425' elevation."

In the VCSNS Database AMR Query table, the applicant identified no aging effect for stainless steel expansion joints, mechanical -bellows, orifices, valves (body only) and pipe and fittings (thermowells) in a sheltered environment. The staff also noted that the applicant identified loss of materials from MIC as an applicable aging effect for stainless steel tube and tube fittings in a sheltered environment. Clarify the applicability of the discussion in VCSNS Database AMR Query Notes (A-SW-f) quoted above to justify the different conclusion for the components mentioned above. In particular clarify which components mentioned above are above or below the 425' elevation and provide the basis for not including MIC as an applicable aging mechanism for the aging effect of loss of materials.

The applicant also does not identify any aging effect for stainless steel tube and tube fittings, valves (body only) in the reactor building environment. Provide the justification. If insignificant concentration of contaminants is part of the justification provide the acceptance criterion and the verification/inspection activities on susceptible locations justifying this basis.

RAI 3.3.2.4.21-4

The applicant identifies no applicable aging effect for carbon steel components in an embedded environment. Provide the specification of the embedded environment. If this environment involves concrete, corrosion of carbon steel components embedded in concrete through carbonation etc. are commonly known degradation processes. Provide the basis for the concluding that no applicable aging effect for carbon steel components in this particular embedded environment.

3.3.2.4.22 Spent Fuel Cooling System

RAI 3.3.2.4.22 -1
In page 211 of the VCSNS Database AMR Query Notes, the applicant states that loss of material due to MIC is identified as an aging effect for vulnerable stainless steel components including pipe and tubing exposed to sheltered environment. Loss of material due to MIC is not identified as aging effect for stainless steel components other than pipe and tubing. Since the spent fuel cooling system has many stainless steel components other than pipe and tubing such as heat exchangers, orifices, pumps, and valves. Provide justification why loss of material due to MIC is identified as an aging effect only for stainless steel pipe and tubing components and not for other stainless steel components such as heat exchangers, orifices, pumps, and valves.

3.3.2.4.23 Thermal Regeneration System

RAI 3.3.2.4.23-1
In the VCSNS Database AMR Query table, the applicant identified only the stainless steel pipe and fittings in the sheltered environment are subject to aging effect of loss of material due to MIC. The rest of the stainless steel components in the same environment in this system are identified as not subject to loss of material due to MIC. Explain why the conclusions are different for the same combination of material and environment.

<u>B.1.26 Preventative Maintenance Activities - Ventilation Systems Inspections</u>

RAI B.1.26-1
Under *Monitoring and Trending*, the LRA states that “routine periodic visual inspections are conducted…in order to detect age-related degradation and to initiate corrective actions as necessary.” Please specify the frequency of these periodic inspections and how the inspection frequency is determined.

RAI B.1.26-2
Under *Monitoring and Trending*, the LRA states that temperatures are trended for the reactor building cooling units (RBCUs). However, *Parameters Monitored or Inspected* and *Detection of Aging Effects* discuss visual inspections and do not mention temperature monitoring. Please clarify how the temperature measurements are used in this program.

<u>B.2.2 Diesel Generator Systems Inspection</u>

RAI B.2.2-1
Under *Operating Experience*, the LRA states that this is a new one-time inspection program for which there is no operating experience; however, plant operating experience should provide information on degradation due to loss of material caused by general corrosion and alternate wetting and drying. Please clarify the operating experience. Identify any degraded conditions of systems within scope of the program that have been experienced (if no degraded conditions have been experienced, so indicate).

B.2.3 Liquid Waste System Inspection

RAI B.2.3-1
The Liquid Waste System Inspection is a one-time inspection program with commitments to follow-up actions based on engineering evaluation of inspection results. This a reasonable approach. However, the applicant has stated that the liquid waste processing system components are exposed to unmonitored and uncontrolled borated water, and that the system is used frequently. In addition, this is a new program for which there is no operating experience. There is a potential for high concentrations of impurities in the water, and the condition of the system is unknown. For these reasons, the staff is concerned with the adequacy of the AMP for managing the aging effects of the components with this combination of material and environment and the lack of operating experience. In light of the above, justify the use of a one-time inspection for the liquid waste system components.

RAI B.2.3-2
Under *Operating Experience*, the LRA states that this is a new one-time inspection program for which there is no operating experience; however, plant operating experience with this system should provide information on any age-related degradation. Please clarify the operating experience with this system. In particular, Provide operating history on the occurrence of crevice, pitting, and stress corrosion cracking in the nuclear plant drains (ND) system and the liquid waste processing system (WL) to justify the use of a one-time inspection for the liquid waste system components.

REQUEST FOR ADDITIONAL INFORMATION
VIRGIL C. SUMMER STATION, UNIT 1
LICENSE RENEWAL APPLICATION
(Sections 4.1, 4.3, 4.5, and 4.6)

4.1 IDENTIFICATION OF TIME-LIMITED AGING ANALYSES

RAI 4.1-1
Table 4.1-1 of the LRA identifies time-limited aging analysis (TLAAs) applicable to Summer. Tables 4.1-2 and 4.1-3 in NUREG-1800 identify potential TLAAs determined from the review of other license renewal applications. The LRA indicates that NUREG-1800 was used as a source to identify potential TLAAs. For those TLAAs listed in Tables 4.1-2 and 4.1-3 of NUREG-1800, that are applicable to PWR facilities and not included in Table 4.1-1 of the LRA, discuss whether there are any calculations or analyses that address these topics at Summer. If calculations or analyses exist that address these topics, discuss how the these calculations or analyses were evaluated against the TLAA definition provided in 10 CFR 50.3.

4.3 METAL FATIGUE

RAI 4.3-1
Section 4.3.1 of the LRA that the transients listed in Table 5.2-2 of the FSAR were used in the design of reactor coolant system components at Summer. Section B3.2.1 of the LRA indicates that thermal fatigue transients have been tracked since operation began at VCSNS. Provide the following information for each of the transients monitored at VCSNS:

(ix) The current number of operating cycles and a description of the method used to determine the number of the design transients from the plant operating history.

(x) The number of operating cycles estimated for 60 years of plant operation and a description of the method used to estimate the number of cycles at 60 years.

(xi) A comparison of the thermal fatigue transients monitored with the transients with the transients listed in Table 5.2-2 of the FSAR. Identify any transients listed in the FSAR that are not monitored by the VCSNS Thermal Fatigue Monitoring Program (TFMP) and explain why it is not necessary to monitor these transients.

RAI 4.3.1-2
The Westinghouse Owners Group issued Topical Report WCAP-14577, Revision 1-A, “Aging Management for Reactor Internals,” to address the aging management of the RVI. The staff review of WCAP-14577, Revision 1-A identified a number of issues that should be addressed on a plant specific basis. Renewal Applicant Action Item 11 specified in WCAP -14577, Revision 1-A indicates that the fatigue TLAA of the reactor vessel internals should be addressed on a plant specific basis. In the LRA, SCE&G indicates that the VCSNS ISI program involves monitoring of thermal transients. List the transients that contribute to the fatigue usage for each component listed in Table 3-3 of WCAP-14577, Revision 1-A and discuss how the ISI program monitors these transients.

RAI 4.3.1-3
The Westinghouse Owners Group issued Topical Report WCAP-14575-A, “Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components,” to address aging management of the RCS piping. Tables 3-2 through 3-16 of WCAP-14575-A list RCS components where fatigue is considered significant. The staff review of WCAP-14575-A identified a number of issues that should be addressed on a plant specific basis. Renewal Applicant Action Item 8 requests that the applicant to address components labeled I-M and I-RA in Tables 3-2 through 3-16 of WCAP-14575-A. In the LRA, SCE&G indicates that the VCSNS ISI program involves monitoring of thermal transients. Discuss how the ISI program addresses the components labeled I-M and I-RA in Tables 3-2 through 3-16 of WCAP-14575-A.

RAI 4.3.1-4
The Westinghouse Owners Group has issued the generic Topical Report WCAP-14574-A to address aging management of pressurizers. The staff review of WCAP-14574-A identified a number of issues that should be addressed on a plant specific basis. Renewal Applicant Action Item 1 requests that the applicant demonstrate that the pressurizer sub-component CUFs remain below 1.0 for the period of extended operation. Table 2-10 of WCAP-14574-A indicates that the ASME Section III Class 1 fatigue CUF criterion could be exceed at several pressurizer sub-component locations during the period of extended operation. WCAP-14574-A also identified recent unanticipated transients that were not considered in the original ASME Section III Class 1 fatigue analyses, including inflow/outflow thermal transients. Provide the following information:

(xii) Confirm that the additional transients discussed in WCAP-14574-A, not considered in the original design, have been addressed at Summer.

(xiii) Show the ASME Section III Class 1 CLB CUFs for the applicable sub-components of the Summer pressurizers specified in Table 2-10 of WCAP-14574-A and the corresponding CUFs for the extended period of operation.

(xiv) Discuss the impact of the environmental fatigue correlations provided in NUREG/CR-6583, “Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels,” and NUREG/CR-5704, “Effects of LWR Coolant Environments on Fatigue on Fatigue Design Curves of Austenitic Stainless Steels,” on the above results.

RAI 4.3.1-5
Section 4.3.1 of the LRA discusses SCE&G’s of the TFMP. The discussion indicates that the program is equivalent to the program described in Section X.M1 of NUREG-1801. The discussion also indicates that the program will be enhanced to incorporate new guidance in EPRI Report, “Materials Reliability Program Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application (MRP-47).” EPRI Report MRP-47 was submitted to the staff for review by NEI letter dated July 31, 2001. By letter dated November 15, 2002, NEI requested that the staff place the review of EPRI Report MRP-47 on hold. As a consequence, the staff has not endorsed the guidelines in EPRI MRP-47. In order to meet the program described in NUREG-1801, the evaluation the reactor water environmental effects should address the fatigue sensitive component locations identified in NUREG/CR-6260, “Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components.”

Provide the following additional information regarding the evaluation of reactor water environmental effects:

1. Confirm that the environmental fatigue correlations contained in NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels," and NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue on Fatigue Design Curves of Austenitic Stainless Steels," will be used in the evaluation.

2. Describe any enhancements to the TFMP resulting from the guidance provided in EPRI Report MRP-47 and provide the technical justification for these enhancements.

3. Provide the design basis usage factors for each of the six component locations listed in NUREG/CR-6260. Identify the transients that are significant contributors to the CUF at these locations.

RAI 4.3.2-1

Section 4.3.2 of the LRA addresses ASME Section III, Class 2 and 3 piping fatigue. The LRA indicates that the post-accident and nuclear sampling systems at Summer could approach the 7,000 cycle limit during the period of extended operation. Provide the material, the maximum calculated stress range, and the allowable stress limit at the bounding location for each of these systems.

4.5 CONCRETE CONTAINMENT (REACTOR BUILDING) TENDON PRESTRESS ANALYSIS

RAI 4.5-1

Section 4.5 of the LRA indicates that the reactor building tendons are a TLAA, and VCSNS will utilize 10 CFR 54.21(c)(1) - Option (iii) to demonstrate that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation. Appendix B.3.3 of the LRA indicates that the Tendon Surveillance Program is consistent with X.S1, Concrete Containment Tendon Prestress, as identified in NUREG-1801. In order for the staff to determine the adequacy of the tendon prestressing force and the TLAA for the period of extended operation, an understanding of the past operating experience for the tendons is needed.

Test results from the first three surveillances indicated that the wire relaxation force losses in the tendon system were greater than the force losses predicted during design (resulting in lower measured prestressing forces). Therefore, in June 1988, the predicted wire relaxation force losses were increased from 8.5% to 12.5%. Then in the fourth period (10th year) tendon surveillance, the vertical tendons were retensioned because the previous surveillance data indicated that the vertical tendon forces would be below the Technical Specifications minimum prior to the fifth period surveillance. Although the fifth period (15th year) and sixth period (20th year) tendon surveillances have been completed, no information was provided regarding the comparison of the measured tendon forces to the predicted lower limit at the 15th and 20th year tendon surveillances. LRA Section 4.5 indicates that based on trending data and results from previous surveillances, "VCSNS does not currently expect the tendons to provide adequate prestress for 60 years without future retensioning of various members."

In order to make a reasonable assessment regarding the effectiveness of the TLAA, the staff requests that the applicant provide the following information:

a. Based on the measurements collected to date, provide the plots of the measured lift-off forces and trend lines along with the predicted lower limits and minimum required values for the three sets of tendons (vertical, horizontal, and dome). These curves should reflect the past retensioning of the tendons. Identify whether the guidance in Information Notice 99-10 is implemented.

b. Provide a brief discussion regarding the reason why the tendon wire relaxation values were greater than those used in the design of the tendon system. Are there any unique characteristics of the Summer tendons or containment design that would cause this to occur. If known, describe operating experience at other plants where similar tendon behavior has occurred.

4.6 CONTAINMENT (REACTOR BUILDING) LINER PLATE, METAL CONTAINMENTS, AND PENETRATION FATIGUE ANALYSIS

RAI 4.6.1
The description of penetrations in Subsection 4.6.3.1 of the LRA indicates that the hot penetrations are sealed on the inside of the containment by a flat plate, welded to both the sleeve and the process pipe at each end of the penetration sleeve. The penetration sleeve is presumably welded to the liner. Provide justification for not evaluating the effects of hot process pipe thermal operating transients and other cyclic loads on potential fatigue of the liner, the hot penetrations, and the process piping at these locations.

REQUEST FOR ADDITIONAL INFORMATION RELATED TO AGING MANAGEMENT PROGRAMS -VIRGIL C SUMMER- LICENSE RENEWAL APPLICATION

AMP B.1.9 Service Water System Reliability and In-Service Testing Program

No RAIs See inspection items

AMP B.1.19 Material Handling System Inspection Program

RAI B.1.19-1
The staff's position, as described in GALL Vol. 2 item VII.B.2-a, is that loss of material due to wear on crane rails falls within the scope of license renewal, even though it is caused by active components. The crane rails are passive, long-lived components, and loss of material due to wear is an applicable aging effect. Provide justification for concluding that loss of material due to wear does not require aging management for VCSNS cranes.

RAI B.1.19-2
The LRA is not clear which cranes are covered by this AMP. The only reference to this AMP is from AMR Table 3.3-1, Item 15; however, there are no LRA Section 2 tables that refer to this AMR item. Please clarify the AMR of the cranes, and clarify which cranes use the Material Handling System Inspection Program.

AMP B.1.25 Preventive Maintenance Activities-Terry Turbine

RAI B.1.25-1
The Preventive Maintenance Activities-Terry Turbine description states in LRA Section B.1.25, under element 5 ("Monitoring and Trending"), that "routine periodic visual inspections are conducted…in order to detect age-related degradation and to initiate corrective actions as necessary." Please specify the frequency of these periodic inspections or how the inspection frequency is determined.

B.1.26 Preventative Maintenance Activities - Ventilation Systems Inspections

RAI B.1.26-1
Under *Monitoring and Trending*, the LRA states that "routine periodic visual inspections are conducted…in order to detect age-related degradation and to initiate corrective actions as necessary." Please specify the frequency of these periodic inspections and how the inspection frequency is determined.

RAI B.1.26-2
Under *Monitoring and Trending*, the LRA states that temperatures are trended for the reactor building cooling units (RBCUs). However, *Parameters Monitored or Inspected* and *Detection of Aging Effects* discuss visual inspections and do not mention temperature monitoring. Please clarify how the temperature measurements are used in this program.

B.2.2 Diesel Generator Systems Inspection

RAI B.2.2-1
Under *Operating Experience*, the LRA states that this is a new one-time inspection program for which there is no operating experience; however, plant operating experience should provide information on degradation due to loss of material caused by general corrosion and alternate wetting and drying. Please clarify the operating experience. Identify any degraded conditions of systems within scope of the program that have been experienced (if no degraded conditions have been experienced, so indicate).

B.2.3 Liquid Waste System Inspection

RAI B.2.3-1
The Liquid Waste System Inspection is a one-time inspection program with commitments to follow-up actions based on engineering evaluation of inspection results. This a reasonable approach. However, the applicant has stated that the liquid waste processing system components are exposed to unmonitored and uncontrolled borated water, and that the system is used frequently. In addition, this is a new program for which there is no operating experience. There is a potential for high concentrations of impurities in the water, and the condition of the system is unknown. For these reasons, the staff is concerned with the adequacy of the AMP for managing the aging effects of the components with this combination of material and environment and the lack of operating experience. In light of the above, justify the use of a one-time inspection for the liquid waste system components.

RAI B.2.3-2
Under *Operating Experience*, the LRA states that this is a new one-time inspection program for which there is no operating experience; however, plant operating experience with this system should provide information on any age-related degradation. Please clarify the operating experience with this system. In particular, Provide operating history on the occurrence of crevice, pitting, and stress corrosion cracking in the nuclear plant drains (ND) system and the liquid waste processing system (WL) to justify the use of a one-time inspection for the liquid waste system components.

B.2.5 REACTOR BUILDING COOLING UNIT INSPECTION

RAI B.2.5-1
The Reactor Building Cooling Unit Inspection program description in LRA Section B.2.5, Element 3, *Parameters Monitored or Inspected*, states that the parameters inspected include visual evidence of loss of material, cracking, or other age-related degradation. Explain how visual inspection can provide information about cracking at the inside surface of piping.

RAI B.2.5-2
LRA Section B.2.5, Element 5, *Monitoring and Trending*, states that no actions are taken as a part of the reactor building cooling unit inspection to trend inspection results. The NRC staff notes that the evaluation of appropriateness of the techniques and timing of the one-time inspection improve with the accumulation of plant-specific and industry-wide experience. As a result of the insights gained from the recent discovery of boric acid-induced corrosion of the Davis-Besse vessel, address the changes that may be made in monitoring and trending (considering that certain components, although stainless steel, are exposed to unmonitored

borated water environment) in response to the Davis-Besse event. Clarify that when inspection results reveal degraded conditions (even from a different system), additional inspections addressed in element 7 ("Corrective Actions") form the basis for future monitoring and trending actions. Also identify to what extent, if any, the boric acid corrosion AMP is integrated with the reactor building cooling unit inspection.

RAI B.2.5-3
LRA Section B.2.5, *Operating Experience*, states that the inspection is a new one-time inspection for which no operating experience exists. Provide operating experience relative to leaks or degradation in the reactor building cooling unit drain piping and drain pan. If no leaks or degradation have been experienced, so indicate.

B.2.6 SERVICE AIR SYSTEM INSPECTION

RAI B.2.6-1
The Service Air System Inspection program description in LRA Section B.2.6, under Element 2, *Preventive and Mitigative Actions*, states that there are no preventive or mitigative actions taken as part of this program. The staff notes that accepted industry guidance and the GALL recommend preventive monitoring of system air quality to ensure that oil, water, rust, dirt, and other contaminants are kept within specified limits. The air quality needs to be maintained because instruments and components may not function properly if the air is contaminated, and the presence of oil or contaminants in the air can impact the rate and types of aging degradation. Describe the monitoring of air quality as it relates to corrosion and degradation of the steel components within the scope of this program.

RAI B.2.6-2
LRA Section B.2.6, *Operating Experience*, states that this program consists of is a new one-time inspection for which no operating experience exists. Discuss the operating experience with the service air system, service air and building services systems, building services system, or instrument air system as it relates of aging degradation of these systems. For example, provide operating experience related to leaks or degradation in the service air system (re: GL 88-14). If no leaks or degradation have been experienced, so indicate.

B.2.8 WASTE GAS SYSTEM INSPECTION

RAI B.2.8-1
LRA Section B.2.8, under *Operating Experience*, states that the inspection is a new one-time inspection for which no operating experience exists. Discuss the operating experience with the gaseous waste processing system as it relates of aging degradation of these systems (such as leaks or degraded conditions related to aging).

B.2.11 INSPECTIONS FOR MECHANICAL COMPONENTS

RAI B.2.11-1
The Inspections for Mechanical Components description in LRA Section B.2.11, under *Program Scope*, states that the relevant aging effect of loss of material is due to galvanic, general, and pitting corrosion. However, the inspections for mechanical components program does not mention MIC, even though the program is credited with the management of loss of

material due to MIC in Table 3.3-1, Item 5. Clarify if the inspections for mechanical components applies to evaluating MIC or if some other aging management program addresses loss of material due to MIC.

RAI B.2.11-2

LRA Section B.2.11, Element 3, *Parameters Monitored or Inspected*, states that the external surfaces of components fabricated of carbon steel, low-alloy steel, and other susceptible materials are inspected for loss of material or cracking. Expand the description of the program to provide the technical basis for the selection of the component external surfaces to be inspected.

For example, are these visual examinations conducted on an opportunistic basis? Are these external surfaces already exposed and accessible to visual examination during normal operation, or do they include external surfaces at susceptible locations that are exposed to visual examination due to targeted planned actions such as equipment disassembly, insulation removal, etc., that may or may not involve suspension of normal operation? If the second group of surfaces is excluded from the AMP, provide the basis. In addition, provide the technical basis for determining how many and what additional component external surfaces are to be inspected if unacceptable degradation is observed in the representative components.

RAI B.2.11-3

Inspections for mechanical components is a new plant specific program with no mention of the qualifications of personnel performing the mechanical inspections. NUREG-1800 section A.1.2.3.6 indicates that qualitative inspections should be performed to same predetermined criteria as quantitative inspections by personnel in accordance with ASME Code and through site specific programs. For example, NUREG-1801 section XI.M.32 for one time inspection indicates that combinations of NDE are performed by qualified personnel following procedures consistent with the ASME Code and 10 CFR50 Appendix B. Define the qualifications of inspection personnel.

RAI B.2.11-4

LRA Section B.2.11, Element 5, *Monitoring and Trending*, states that the inspections will be performed and documented in accordance with station procedures and, following baseline inspection, the frequency of inspections will be determined based on inspection results and industry experience. Provide the schedule for the baseline inspection.

Editorial Comment: LRA Section B.2.11, Element 7, *Corrective Actions*, states, "If the results of the inspections for mechanical components are not acceptable, as determined by the engineering evaluation, then corrective actions are taken to repair or replace the ***effective*** components." Should this read "affected" components?

RAI B.2.11-5

LRA Section B.2.11, under *Operating Experience*, states that the inspection is a new inspection. The applicant states that there is VCSNS relevant operating experience with the identification of pitting below the insulation in the chilled water system, which was detected and repaired under existing inspection activities, and that several instances of leakage in the chilled water system have been identified by surveillance procedures. Discuss any additional related operating experience relevant to the systems within scope or confirm that this is the only system in the scope of this program with observed degraded conditions.

RAI B.2-11-6
LRA Table 3.3-1, Item 5 credits the Inspections for Mechanical Components program for managing loss of material of the chilled water expansion tanks. GALL Program XI.M29 addresses aboveground carbon steel tanks including inaccessible areas, but the LRA does not include this program. Describe how the Inspections for Mechanical Components program addresses aboveground carbon steel tanks, including inaccessible locations, and other elements addressed in XI.M29.

RAI B.2-11-7
The AMP 2.11, Inspection for Mechanical Components, a new plant specific program, with no NUREG-1801 parallel, is credited for managing loss of material due to general corrosion and crack initiation and growth due to cyclic loading and SCC of the carbon and alloy steel component/component types and inherently addresses their closure bolting in the auxiliary (AS) and the steam and power conversion (SPC) systems. In Table 3.2-1, AMR Item12 (ESF); Table 3.3-1, AMR Item 23 (AS); and in Table 3.4-1, AMR Item 8 (SPC); the applicant states that the specific bolting/fasteners materials within the scope of license renewal were not itemized as a separate Non-Class 1 component/component types. Rather, bolting was treated as "piece-part" (or sub-component/sub-part) of Non-Class 1 components/component types.

The staff notes that NUREG-1801 credits AMP XI.M 18 Bolting Integrity for monitoring loss of material, cracking, and loss of preload. In addition, accepted bolting integrity programs (such as EPRI 104213) recommend monitoring for loss of preload as one of the parameters monitored/inspected. Monitoring for cracking of high strength bolts (actual yield strength equal or greater than 150 ksi) is also recommended.

As such, the applicant is requested to provide the following information:
- Identify the AMP that will manage the aging effects for ESF closure bolting (Table 3.2-1, AMR Item 12).
- Justify how the AMPs credited in the VCSNS LRA for bolting are consistent with the Bolting Integrity AMP.
- Provide justification for concluding that loss of preload is not an applicable aging effect.
- Are there any high strength bolts included within the boundary of these three systems (Engineered Safety Features, Auxiliary, and Steam & Power Conversion Systems)?

B.2.12 HEAT EXCHANGER INSPECTIONS

RAI B.2.12-1
The Heat Exchanger Inspections (HEI) program is credited in LRA Section B.2.12 with detecting and characterizing loss of material due to selective leaching and erosion-corrosion, as well as heat exchanger fouling due to particulates, for heat exchanger components in a treated water environment. Provide information regarding management of galvanic corrosion of heat exchanger tubes.

RAI B.2.12-2
LRA Section B.2.12 states that the HEI program is a one-time inspection. For all heat exchanger components in the component cooling water system subject to aging effects for

which the Chemistry Program (CP) and the HEI are applicable AMPs, discuss whether both AMPs are used together to manage all applicable aging effects. The LRA is unclear on this point because the CP explicitly exempts one-time inspection, but the LRA states that HEI is consistent with GALL Programs XI.M32 and XI.M33. Discuss whether the HEI is used to verify the effectiveness of the CP for the applicable aging effects.

RAI B.2.12-3
LRA Section B.2.12, Element 4, *Detection of Aging Effects*, states that a combination of proven volumetric and visual examination techniques will be used at sample locations in the various heat exchangers determined by engineering evaluation to be most susceptible to the applicable aging effects. The LRA states that if no parameters are known that would distinguish the susceptible locations, sample locations will be selected based on accessibility and radiological concerns, and the results will be applied to the associated components. Discuss how the results of sampling would be taken into account for any future inspections (monitoring).

RAI B.2.12-4
LRA Section B.2.12, Element 6, *Acceptance Criteria*, states that the acceptance criteria are no unacceptable loss of material or heat exchanger fouling that could result in a loss of the component intended function(s) as determined by engineering evaluation. Elaborate on the acceptance criteria applied in the engineering evaluation and explain how a determination of no unacceptable loss of material or cracking of subject components can be made on the basis of a one-time inspection with consideration to the rate of damage.

RAI B.2.12-5
LRA Section B.2.12, *Operating Experience*, states that this is a new one-time inspection for which no operating experience exists. Comment on any relevant operating experience for the systems that will be managed by this program.